

Korean Assessment of the Proliferation Resistance on the Whole Fuel Cycle of DUPIC

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Myung Seung YANG, Joo Hwan PARK

**Korea Atomic Energy Research Institute
Republic of Korea**

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1. Introduction

IAEA (International Atomic Energy Agency) has begun the INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) program in 2000 following the recommendation of the 44th General Conference. INPRO proposed the proliferation resistance as a key component of the future innovative nuclear system for fulfilling the energy needs in the 21st century along with sustainability, economics, safety of nuclear installation and waste management. The Republic of Korea is participating in the INPRO program from the beginning.

In order to set out desired goals of the innovations of the nuclear energy system, a set of BP(Basic Principles), UR(User Requirements) and Criteria including Indicators and Acceptance Limits were developed and published as the IAEA-TECDOC-1362 (Guidance for the evaluation of innovative nuclear reactors and fuel cycles) in result of the INPRO Phase 1A study which was performed till June 2003. In the proliferation resistance area, 5 Basic Principles, 5 User Requirements and related Criteria were presented in the IAEA-TECDOC-1362 [1].

From July 2003, INPRO Phase 1B, Part 1 was started to verify the completeness and the adequacy of the INPRO Methodology and to recommend further improvement by applying them for several Case Studies. During this phase, 6 National Case Studies by the Russian Federation, India, Argentina, Korea, China and Czech were performed.

KAERI (Korea Atomic Energy Research Institute) has perform the National Case Study on the DUPIC (Direct use of PWR spent fuel in CANDU reactors) to assess the adequacy of the proposed INPRO Methodology in IAEA-TECDOC-1362 by applying them to the DUPIC in the area of the proliferation resistance as an activity of the INPRO Phase 1B, Part 1 until December 2004.

The Korean Case Study was focused on the application of the INPRO Methodology to mainly DUPIC fuel fabrication activity.

The major findings and recommendation of the Korean Case Study in the area of proliferation resistance were as follows.

- INPRO Methodology is useful but needed further development.
- Lack of link or correspondence between BPs and URs of the INPRO Methodology

- Overlap or redundancy of BPs and URs
- Lack of practical guidance for the application of INPRO methodology for PR (Proliferation Resistance)
- Deleting redundancies and modifying of BPs and URs (It recommends the 2 BPs and 4 URs rather than 5 BPs and 5 URs as was shown in TECDOC-1362.)
- Necessity of inter-relationship between BPs and URs
- Each PR barriers/Indicators should be quantified.
- Development of the aggregation method of PR barriers, Indicators and URs

The deficiencies of the INPRO Methodology of IAEA-TECDOC-1362 in the proliferation resistance area such as the redundancy of the Basic Principles and User Requirements, and the lack of relationship between BPs and URs, etc. were improved based on the Korean Case Study. Korean proposal was further improved by two consultancy meetings and it proposed the revision of the 2 Basic Principles, 5 User Requirements and 7 Indicators. The revised INPRO Methodology was published as the IAEA-TECDOC-1434 (Methodology for the assessment of innovative nuclear reactors and fuel cycles) in December 2004 [2].

As INPRO Phase 1B, Part 2 was started in 2005, KAERI has started the Extended Case Study on the whole DUPIC fuel cycle in the proliferation resistance area as an activity for the INPRO Phase 1B, Part 2 in January 2005, which will be performed till June 2006. The main purpose of the Extended Case Study on the whole DUPIC fuel cycle is to assess the adequacy of the revised INPRO Methodology proposed in IAEA-TECDOC-1434. During this Extended Case Study, further improvement of the INPRO Methodology in the area of proliferation resistance is recommended for the modification, and the modified methodology is applied to the whole DUPIC fuel cycle to evaluate its adequacy. The results of the Korean Extended Case Study on the whole DUPIC fuel cycle are described in this report.

2. Scope of Study

2.1 Objectives

The main objective of this Extended Case Study on the whole DUPIC fuel cycle is: (1) to review the INPRO Methodology in the area of proliferation resistance described in the IAEA-TECDOC-1434, (2) to assess its adequacy and completeness in the light of the PR characteristics of the whole DUPIC fuel cycle encompassing the supply of feed uranium oxide (LEU) from foreign country, PWR fuel fabrication, PWR spent fuel transportation, DUPIC fabrication, DUPIC spent fuel disposal, etc., (3) to suggest, resulting from the assessment, further improvement of the Methodology.

2.2 Steps of the Study

The Extended Case Study on the whole DUPIC fuel cycle is performed in 4 steps as described below.

Step 1: Review of the INPRO Methodology (IAEA-TECDOC-1434) for the application to the DUPIC fuel cycle for the further improvement and modification.

- Applicability of Basic Principle (BP), User Requirement (UR) and Criteria to the DUPIC fuel cycle

Step 2: Establishment of the evaluation framework of the DUPIC fuel cycle

- Establishment of the evaluation stages of DUPIC fuel cycle from the supply of feed uranium oxide (LEU) from foreign country, LEU fuel fabrication, PWR operation, PWR spent fuel handling, DUPIC fuel fabrication, CANDU operation, and DUPIC spent fuel handling to final disposal.
- Establishment of an evaluation model of the DUPIC fuel cycle for the application of INPRO Methodology

Step 3: Assessment of the PR characteristics of the whole DUPIC fuel cycle using the modified INPRO Methodology.

- Characterization of the process flow and facilities for each stage of the DUPIC fuel cycle
- Determination of the PR characteristics of each stage of the DUPIC fuel cycle and the variables to be used for the evaluation.
- Quantification of the PR variables (evaluation parameters) for each Indicator of User Requirements

- Evaluation of the PR Indicators of the DUPIC fuel cycle
- Evaluation of the cost efficiency of PR measures
- Integration of the assessment results of each stage
- Establishment of a model for the aggregation and presentation of evaluation results
- System assessment of PR of the DUPIC fuel cycle
- Uncertainty assessment of the evaluation

Step 4: Recommendation on the further improvement of the INPRO Methodology in the area of the proliferation resistance.

The work procedures and the outline of the schedules of the Extended Case Study are shown in Fig. 1 and Fig. 2, respectively.

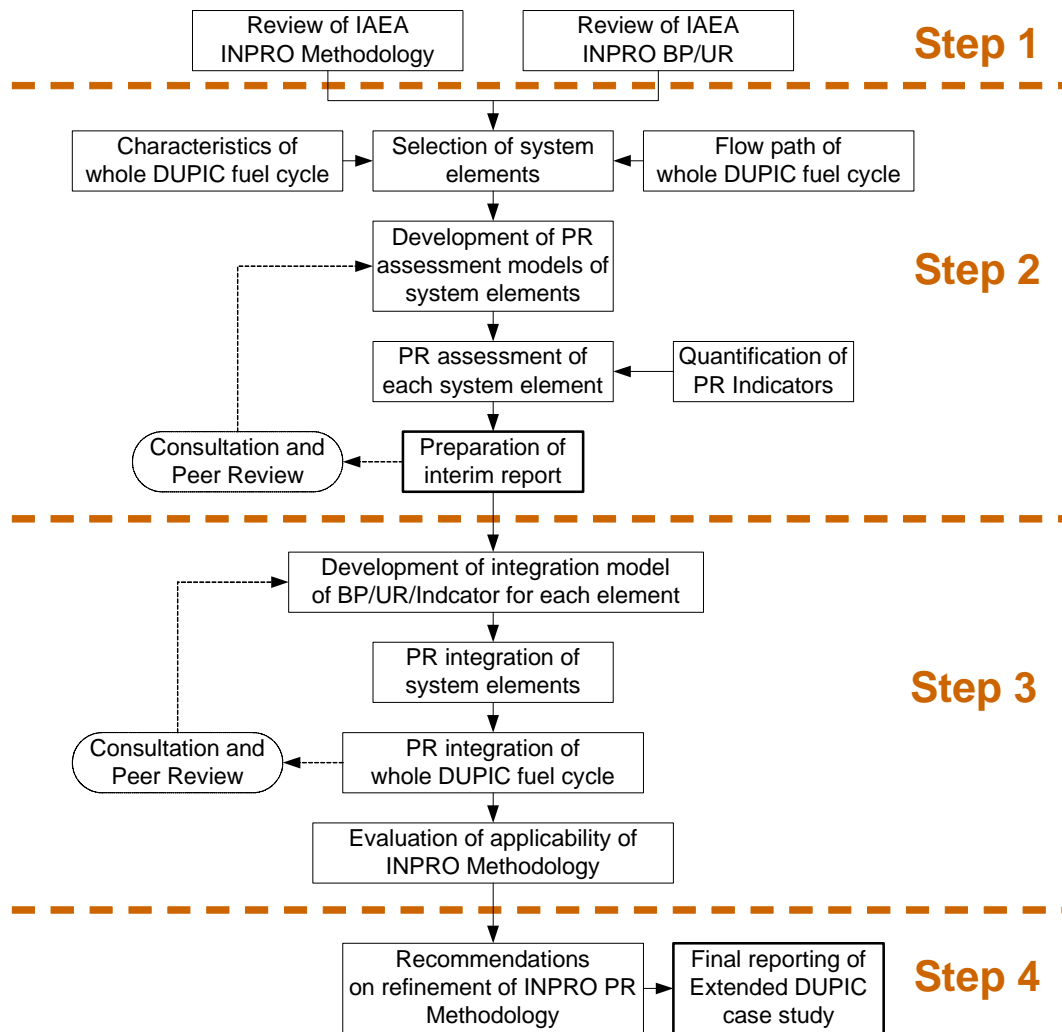


Figure 1. Work procedure of the INPRO Extended Case Study on DUPIC

Work Scope	Period (Month)	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct
Proposal of INPRO DUPIC Case Study									
Review of INPRO Methodology									
- Applicability of BP, UR and Criteria to DUPIC									
Establishment of the evaluation framework									
- Evaluation stages of whole fuel cycle of DUPIC									
- Evaluation model of whole fuel cycle of DUPIC									
Assessment of the PR characteristics of the whole DUPIC cycle									
- Characterization of the process flow and facilities for each stage									
- Determination of PR characteristics and variables of each stage									
- Quantification of the PR variables for each Indicator of UR									
- Evaluation of the PR Indicators									
- Evaluation of the cost efficiency of PR measures									
Integration of the assessment results of each stage									
- Establishment of aggregation model and presentation of results									
- System assessment of PR									
- Uncertainty assessment of the evaluation									
Recommendation on the further improvement									
Documentation									

Figure 2. Work schedule of the INPRO Extended Case Study on DUPIC

3. DUPIC Fuel Cycle

3.1 The concept of DUPIC fuel cycle

The basic concept of DUPIC fuel cycle is to fabricate the CANDU nuclear fuel from the PWR spent fuel by use of dry thermal/mechanical processes without separating stable fission products. Since the CANDU reactor utilized the natural uranium fuel, contents of the remained fissile materials in PWR spent fuel is large enough to be reused in CANDU reactor even though it still contains fission products [3].

The advantages of utilizing the DUPIC fuel cycle are: (1) to get rid of the PWR spent fuel, which is to be refabricated to the CANDU fuel, (2) to save natural uranium resources to be required to produce CANDU fuel, and (3) to reduce the spent CANDU fuel accumulation thank to its high burnup. The basic concept of DUPIC fuel cycle is schematically shown in Fig. 3.

The main element of DUPIC fuel cycle is the manufacturing step of the DUPIC fuel from PWR spent fuel. The manufacturing process flow is described in Fig. 3 schematically. The PWR spent fuel is first disassembled and PWR spent fuel elements are extracted from the assembly. The spent fuel elements are cut to small rodcuts for the easy handling. The rodcuts are decladded by mechanical and/or thermal method to retrieve the PWR spent fuel materials. The PWR spent fuel materials are subject to a series of the oxidation and reduction to make them resinterable by the process named OREOX (Oxidation and reduction of oxide fuel). The oxidation and reduction are performed at 450 °C in air and 750 °C in Ar-4%H₂ atmosphere, respectively. During the oxidation and reduction, the about 30 % volume changes make the spent fuel material finer particles and soft materials with lots of the microcracks, that make them resinterable powder.

Once the resinterable powder feedstock is prepared, the followed manufacturing processes are quite similar to the conventional CANDU fuel manufacturing using powder/pellet route. They are composed of the precompaction, granulation, compaction, sintering, grinding, end cap welding by the laser, and final assembling of the DUPIC bundle.

Since there are no process steps for the separation of the fission products and transuranic materials while the volatile and semi-volatile elements are removed during

the thermal/mechanical treatments, the process materials are very radioactive throughout whole manufacturing processes. Therefore, the manufacturing process should be performed inside the heavily shielded hot cell by remote manners. The characteristics incur the difficulties in material handling during manufacturing, but it is an strong incentive in terms of the proliferation resistance of the DUPIC fuel.

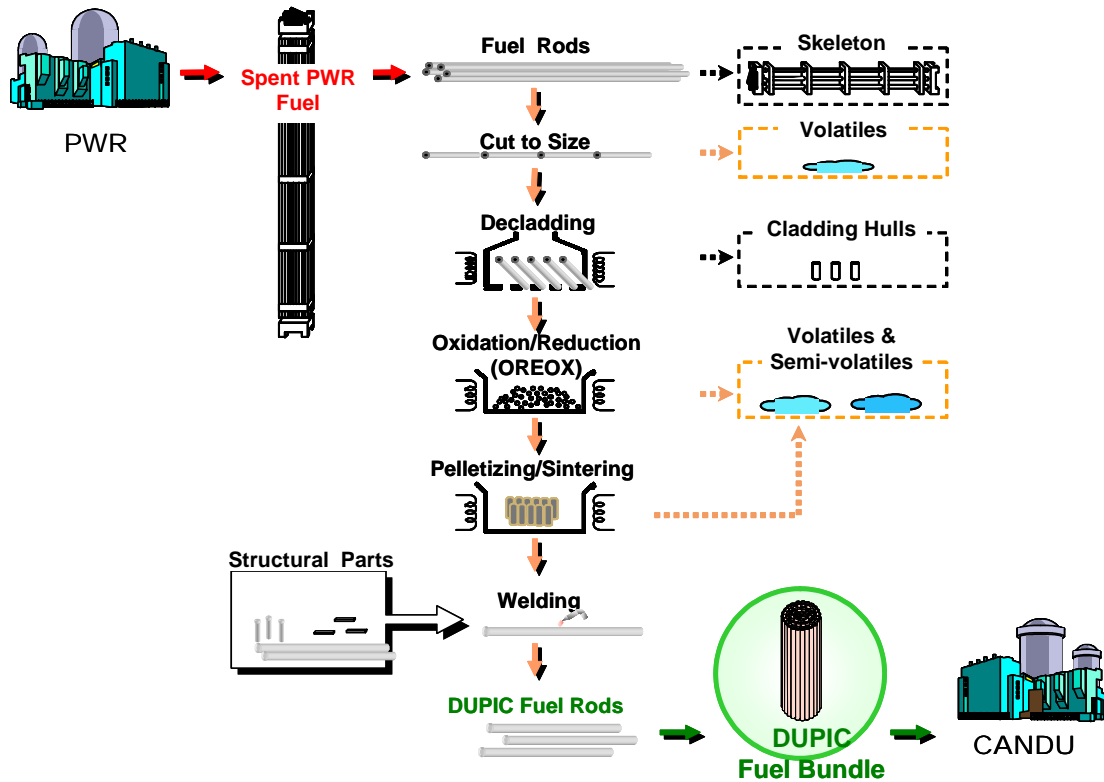


Figure 3. Concept of DUPIC fuel cycle

In order to realize the DUPIC fuel cycle in practice, the remote fabrication technology has to be developed, and the performance and the compatibility of DUPIC fuel have to be demonstrated.

3.2 The whole DUPIC fuel cycle

Since the DUPIC is the synergism between PWR and CANDU fuel cycle, the whole DUPIC fuel cycle, in the Korean case, is beginning from the supply of feed uranium from a foreign country for the PWR fuel production to the final DUPIC spent fuel disposal. The system elements of the whole DUPIC fuel cycle are: supply of feed uranium from a foreign country, fabrication of PWR fuel, burning in PWR, discharge of PWR spent fuel, interim storage of PWR spent fuel, DUPIC fuel fabrication, burning of

DUPIC fuel in CANDU, discharge of DUPIC spent fuel, interim storage of DUPIC spent fuel and final disposal of DUPIC spent fuel.

The system elements of the whole DUPIC fuel cycle are show in Fig. 4 and Fig. 5.

<u>PWR fuel cycle</u>	<u>DUPIC fuel cycle</u>
· Step P1: Supply of feed uranium from a foreign country for the PWR fuel	· Step D1: Transportation of PWR spent fuel to DUPIC fabrication facility
· Step P2: Transportation of LEU material to PWR fuel fabrication facility	· Step D2: DUPIC fabrication plant
· Step P3: PWR fuel fabrication facility	· Step D3: Transportation of DUPIC fuel to CANDU plant
· Step P4: Transportation of PWR fuel to PWR plant	· Step D4: CANDU plant
· Step P5: PWR plant	· Step D5: Transportation of DUPIC spent fuel to interim storage
· Step P6: Transportation of PWR spent fuel to interim storage	· Step D6: Interim storage of DUPIC spent fuel
· Step P7: Interim storage of PWR spent fuel	· Step D7: Transportation of DUPIC spent fuel to permanent disposal
	· Step D8: Permanent disposal of DUPIC spent fuel

Figure 4. System elements of the whole DUPIC fuel cycle

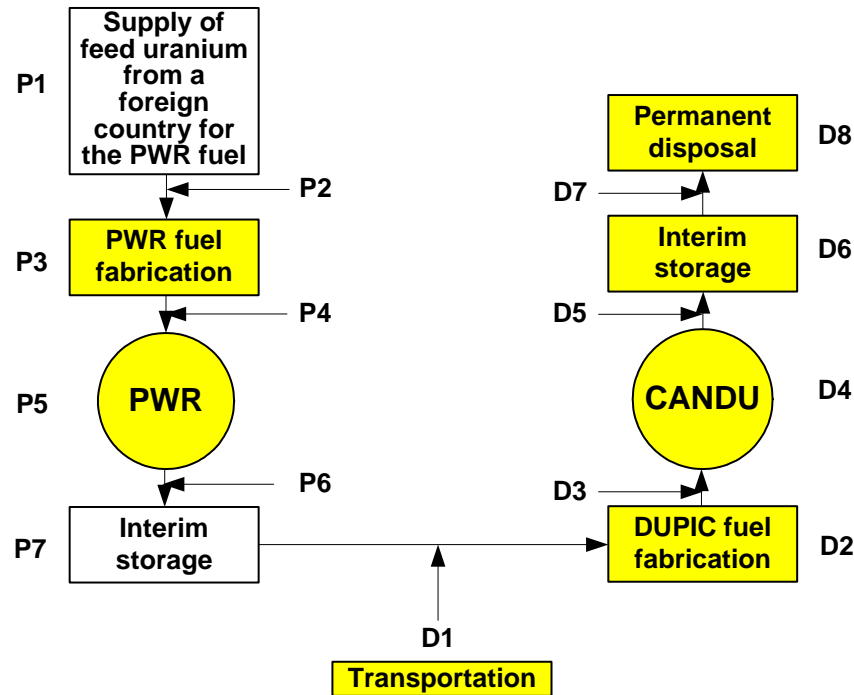


Figure 5. System flow of the whole DUPIC fuel cycle

In addition, the transportation of the nuclear materials among the system elements is also important step to be considered in the viewpoint of the proliferation resistance. The system elements of the whole DUPIC fuel cycle are schematically shown in Fig. 5.

In performing the extended case study of the whole DUPIC fuel cycle in the area of the proliferation resistance, the system elements for the production of feed uranium oxide (from the mining to the enrichment of LEU) are not considered here, because the Republic of Korea does not perform activities from the mining to enrichment at present, and import all the enriched uranium oxide powder from foreign countries. Therefore, the process characteristics from the PWR fuel fabrication to the final disposal of the DUPIC spent fuel are evaluated for the proliferation resistance feature of the whole DUPIC fuel cycle using INPRO Methodology.

4. Review of the INPRO Methodology in IAEA-TECDOC-1434

The outline of the INPRO Methodology in the proliferation resistance area described in the IAEA-TECDOC-1362 is shown in the Fig. 6. In IAEA-TECDOC-1362, there are 5 Basic Principles and 5 User Requirements, which were reviewed through the various Case Studies including DUPIC in the INPRO Phase1B, Part 1 from July 2003 to December 2004. As the Case Studies pointed out, they needed some improvements because of the redundancy of the Basic Principles and User Requirements, and the lack of the correspondence between Basic Principle and User Requirement.

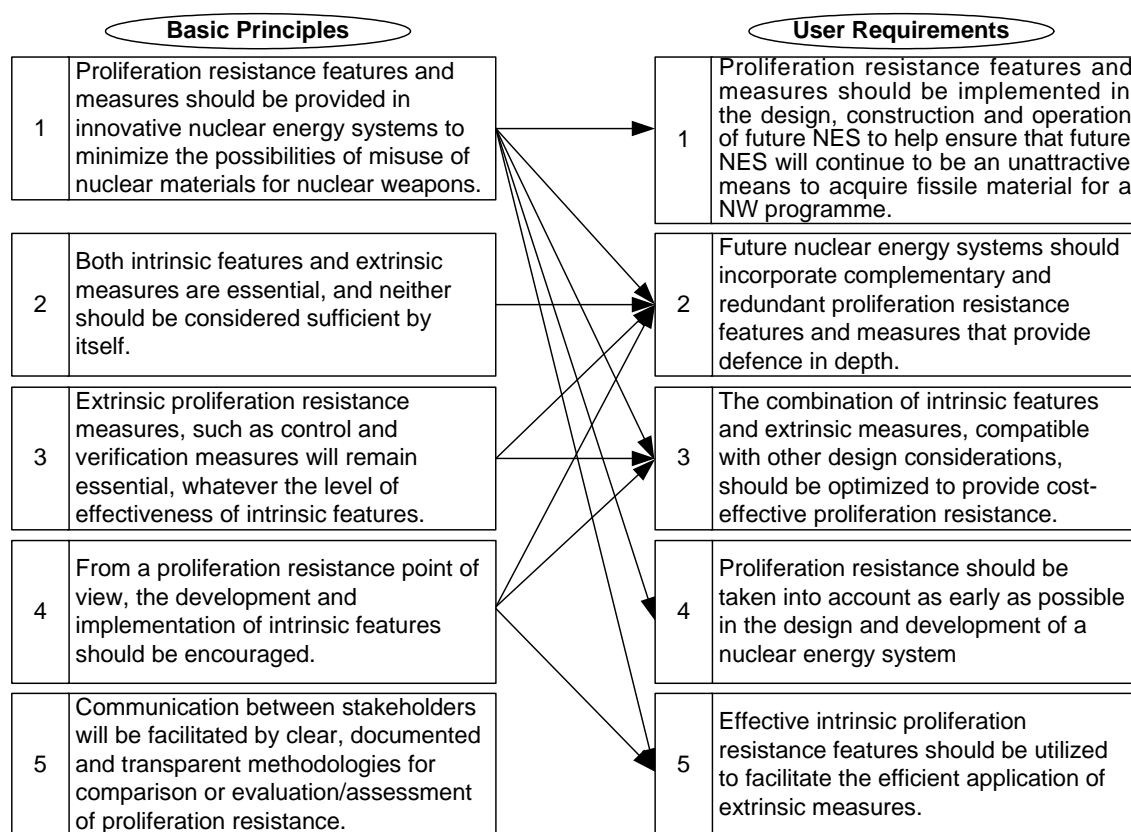


Figure 6. Hierarchy of INPRO Methodology in proliferation resistance of IAEA-TECDOC-1362 [1]

It was revised based on the results of the various Case Studies, and published as an IAEA-TECDOC-1434 in December 2004. According to Chapter 8 of IAEA-TECDOC-1434, in the area of proliferation resistance, two BPs, five URs and seven Indicators with Criteria and Acceptance Limits had been established. Fig. 7 shows the details of the system as described in IAEA-TECDOC-1434.

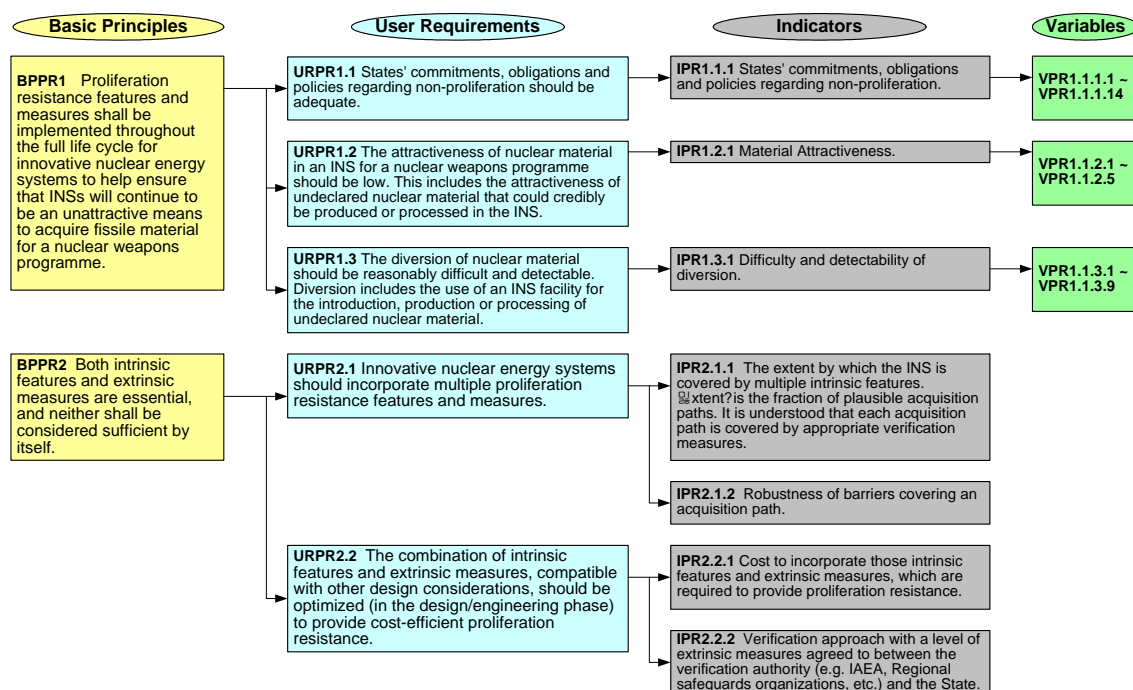


Figure 7. Hierarchy of INPRO Methodology in proliferation resistance of IAEA-TECDOC-1434 [2]

In order to evaluate the proliferation resistance of a nuclear energy system more objectively, a set of Indicators, Criteria and Acceptance Limits corresponding to each User Requirement are prepared. The details of the Indicators in the IAEA-TECDOC-1434 are shown in Tables 1 and 2.

Table 1. Indicators of BPPR1

BPPR1 Proliferation resistance features and measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that INSs will continue to be an unattractive means to acquire fissile material for a nuclear weapons programme.	
URPR1.1 States' commitments, obligations and policies regarding non-proliferation should be adequate.	IPR1.1.1 States' commitments, obligations and policies regarding non-proliferation.
URPR1.2 The attractiveness of nuclear material in an INS for a nuclear weapons programme should be low. This includes the attractiveness of undeclared nuclear material that could credibly be produced or processed in the INS.	IPR1.2.1 Material Attractiveness.
URPR1.3 The diversion of nuclear material should be reasonably difficult and detectable. Diversion includes the use of an INS facility for the introduction, production or processing of undeclared nuclear material.	IPR1.3.1 Difficulty and detectability of diversion.

Table 2. Indicators of BPPR2

BPPR1 Proliferation resistance features and measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that INSs will continue to be an unattractive means to acquire fissile material for a nuclear weapons programme.	
URPR2.1 Innovative nuclear energy systems should incorporate multiple proliferation resistance features and measures.	IPR2.1.1 The extent by which the INS is covered by multiple intrinsic features. “Extent” is the fraction of plausible acquisition paths. It is understood that each acquisition path is covered by appropriate verification measures. IPR2.1.2 Robustness of barriers covering an acquisition path.
URPR2.2 The combination of intrinsic features and extrinsic measures, compatible with other design considerations, should be optimized (in the design/engineering phase) to provide cost-efficient proliferation resistance.	IPR2.2.1 Cost to incorporate those intrinsic features and extrinsic measures, which are required to provide proliferation resistance. IPR2.2.2 Verification approach with a level of extrinsic measures agreed to between the verification authority (e.g. IAEA, Regional safeguards organizations, etc.) and the State.

The main characteristics of the intrinsic features and extrinsic measures are also explained in the TECDOC. Proliferation resistance can be assured by the combination of the intrinsic features and extrinsic measures. **Four type of intrinsic feature** are considered in the IAEA-TECDOC-1434, which is described as follows.

- **First Type**

It consists of the technical features of a nuclear energy system that reduce the attractiveness for nuclear weapons programmes of nuclear material during production, use, transport, storage and disposal. (e.g., Isotope content, Chemical form, Radiation field, Heat generation, Spontaneous neutron generation rate)

- **Second Type**

It comprises the technical features of a nuclear energy system that prevent or inhibit the diversion of nuclear material. (e.g., Design features that limit access to NM, Effectiveness of prevention of diversion of NM, Time required to divert or produce NM and convert it to weapons useable form, Bulk and mass)

- **Third Type**

It consists of the technical features of a nuclear energy system that prevent or inhibit the undeclared production of direct-use material. (e.g., Complexity of and time required for modifications necessary to use a civilian INS for a

weapons production facility, Skills, expertise and knowledge required to divert or produce NM and convert it to weapons useable form, Difficulty to modify fuel cycle facilities and process for undeclared production)

- **Fourth Type**

It consists of the technical features of a nuclear energy system that facilitate verification, including production continuity of knowledge. (e.g., Diversion detectability, Material stocks and flows)

In the IAEA-TECDOC-1434, **five categories of extrinsic measures** are considered, which are described as follows.

- **First Category**

It is States' commitments, obligations and policies with regard to nuclear non-proliferation. These include the NPT and nuclear-weapons-free zone treaties, comprehensive IAEA safeguards agreements and protocols additional to such agreements. (e.g., Safeguards agreements pursuant to the NPT, Nuclear-weapons-free zone treaties, Comprehensive IAEA safeguards agreements, Additional protocols of IAEA agreements)

- **Second Category**

It consists of agreements between exporting and importing States that nuclear energy systems will be used only for agreed purposes and subject to agreed limitations. (e.g., Export control policies, Bi-lateral agreements for supply and return of nuclear material, Bi-lateral agreements governing re-export of NES components)

- **Third Category**

It consists of commercial, legal or institutional arrangements that control access to nuclear material and nuclear energy systems. This can include use of multi-national fuel cycle facilities, and arrangements for spent fuel take-back. (e.g., Commercial, legal or institutional arrangements that control access to NM and NES, Relevant international conventions, Multi-lateral ownership, management or control of a NES)

- **Fourth Category**

It is application of IAEA verification and, as appropriate, regional, bilateral and national measures, to ensure that States and facility operators comply with non-proliferation or peaceful-use undertakings. (e.g., Verification activities, State or regional systems for accounting and control, Safeguards approaches for the State's or regional safeguard system, capable of detecting diversion or undeclared production)

- **Fifth Category**

It consists of legal and institutional arrangements to address violations of nuclear non-proliferation or peaceful-use undertakings. (e.g., An effectiveness international response mechanism for violations)

Additionally, in assessing the proliferation resistance the **following features** had better be considered.

- High Level Intrinsic Features include Multi-national fuel cycle facilities, Co-location of fuel cycle facilities, Closure of fuel cycles, Stockpiling and Potential significance of source material.
- Centralization can provide stronger international control of proliferation-sensitive enrichment and reprocessing technology.
- Co-location can limit transportation and storage of potentially proliferation-sensitive material.
- Closure of Fuel Cycle, which can minimize the quantity of nuclear material in the fuel cycle and the production of proliferation-sensitive material, provides benefits for proliferation resistance.
- Problems of stock piling or maintaining excessive inventories of nuclear material: Minimizing inventory provides benefits for PR, but fuel cycle with small inventories can provide easiness of undeclared production.
- Source Materials
 - Natural uranium, depleted uranium, and thorium provide input material for many fuel cycles.
 - Although not directly useable in a nuclear weapon, these materials require due consideration in a PR assessment because they can be used as source material to generate weapons usable materials.

The evaluation of the proliferation resistance is more difficult than the evaluation of

other technical areas such as safety and sustainability because of the characteristics explained below.

- Malevolent human activity
 - Other areas compared to PR are primarily concerned with technical aspects such as equipment/system failures, radioactive releases, costs, human health, etc.
 - Whereas in most areas it is assumed that agreements are respected and followed, with proliferation it is assumed that non-proliferation agreements are broken.
- Involvement of the interaction between two sides such as the proliferators and the safeguarder/defender
 - It is sometimes examined using gaming theory.
 - The choices that each side makes depend to some extent on what choices they expect the other side to make.
 - This human element must be considered in making a comprehensive assessment of PR, and is further complicated because many analysts believe that proliferators would disregard common safety and environmental norms.
- Requirement of a means to handle sensitive information without disclosing the sensitive details
 - The detailed understanding of how the nuclear material characteristics (e.g., isotopic composition, chemical composition, etc.) affect a nuclear explosive is generally classified information.
 - This makes assessment of the PR provided by material characteristics difficult when considered in more than a coarse sense (e.g., HEU versus LEU or WG Pu versus RG Pu)
- Inherently qualitative and difficult to quantify many of the elements
 - Some elements, such as treaties, agreements, and policies are difficult to quantify because of variations in strength, quality and degree of compliance (a political judgement).
 - Others are difficult to quantify because they involve human choices and activities that are outside of the range of normal experience.

- The technical difficulty of extracting Pu from irradiated targets can vary considerably depending on what the potential proliferator is prepared to do.
- If human health is not a significant consideration, then extraction can be performed with minimal shielding and protective equipment.

Moreover, **quantitative evaluation of the proliferation resistance** requires further development regarding the tasks as shown below.

- Aggregation method
 - Aggregation methods may be
 - ✓ required an accepted means using clear and transparent tools
 - ✓ useful by verification regimes to assess the effect of verification (extrinsic measures) to provide effective and cost-effective PR for a NES
 - ✓ composite incorporating scenario-based and attribute-based tools
- Aggregation methods may be misleading, possible hiding weak links with a single score for PR based on the strengths and weakness of the methodology.
- Dependent and independent State specific information
 - Dependent State specific information
 - ✓ The strength of the PR provided by some intrinsic features can depend on state-specific information such as, *inter alia*, the presence of indigenous uranium resources or the presence of other nuclear facilities.
 - ✓ State-specific extrinsic measures such as fuel supply agreements for procurement of fresh fuel and return of spent fuel (e.g., commitment to multilateral fuel cycle facilities) can affect the PR of an INS.
 - ✓ Independent State specific information
 - Intrinsic features that facilitate verification generally provide PR independent of the State in which the INS is deployed.
 - PR assessments must address both aspects. Where required, credible stylized state descriptions can provide a means to address the state-specific aspects early in the design process.

When the Korean Extended Case Study on the whole DUPIC fuel cycle is performed,

the INPRO Methodology of the proliferation resistance in the IAEA-TECDOC-1434 is first reviewed for its completeness, and the new modified Indicators are proposed. Then, the new modified methodology is utilized to assess the proliferation resistance characteristics of the whole DUPIC fuel cycle. The details of the new modification of the INPRO Methodology in the proliferation resistance area and the results of its application to the whole DUPIC fuel cycle are described in the following chapters.

5. New Proposed Indicators for Modification of IAEA-TECDOC-1434

5.1 New structure of Basic Principles, User Requirements and and Indicators

As described in the previous Chapter, the Indicators of User Requirements under Basic Principle 1 in IAEA-TECDOC-1434 were set to be one for each UR. Each Indicator is similar to the wording of the corresponding UR, but expressed in the concise words to represent the role of Indicators for URs. The Intrinsic features and Extrinsic measures which are most important barriers of proliferation resistance are expressed as variables under the corresponding Indicators.

However, it is desirable that the Indicator itself be considered as the measures of the technical barriers and have its own meaningful characteristics regarding to PR. Hence, the new modified structure of BPs and URs including Indicators which are rearranged for the improvement is shown in Figure 8.

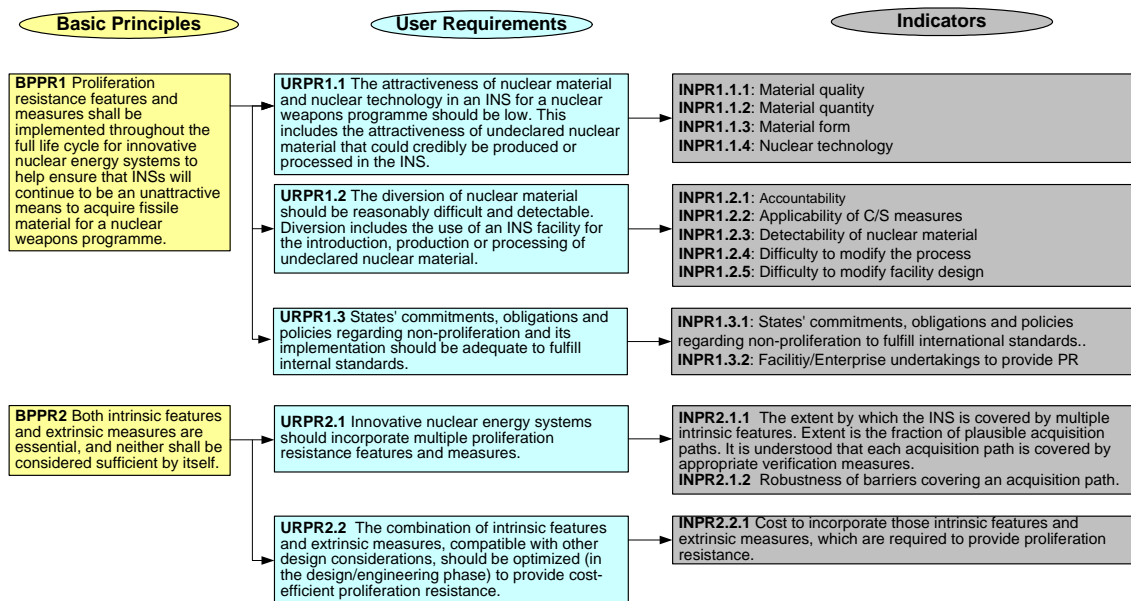


Figure 8. New modified structure of BPs, URs and Indicators in INPRO PR area

The outline of the modification of the IAEA-TECDOC-1434 is summarized below.

The modified URPR1.1 is come from the previous URPR1.2 of IAEA-TECDOC-1434. Moreover, the “Variables” in TECDOC-1434 are rearranged and four new Indicators of URPR1.1 are proposed.

Moreover, regarding the modified URPR1.1, “Nuclear Tecnology” is added to the previous URPR1.2 of IAEA-TECDOC-1434, that was relocated as URPR1.1, because the nuclear technologies such as possession of enrichment facility, technology capability of extraction of fissile material and irradiation capability of target by reactor or accelerator is directly linked with the meaning of the phrase of “Attractiveness of undeclared nuclear material that could credibly produced or processed in the INS for a nuclear weapons programme should be low”. And four new Indicators are proposed to classify the detailed evaluation parameters, which are important to evaluate the Intrinsic barriers regarding material characteristics and nuclear technology.

URPR1.2 is come from the previous URPR1.3 and is same as User Requirement of the previous URPR1.3 of IAEA-TECDOC-1434. However, the five new indicators are proposed to classify the meaning of the evaluation parameters and variables given in IAEA-TECDOC-1434.

The URPR1.3 is come from the previous URPR1.1, and has two Indicators to represent the State’s situation instead of one Indicator in IAEA-TECDOC-1434. Modified INPR1.3.1 is mainly come from the previous INPR1.1.1 of IAEA-TECDOC-1434. Moreover, modified INPR1.3.1 is including the previous INPR2.2.2 of IAEA-TECDOC-1434 as one of the evaluation parameters.

INPR1.3.2 is newly proposed to emphasize the facility/enterprise undertakings such as multi-lateral ownership, which was considered as part of previous INPR1.1.1.

Regarding URPR2.1 of BPPR2, it is not modified at all.

Regarding URPR2.2, INPR2.2.1 is not modified at all. The previous Indicator, INPR2.2.2 in IAEA-TECDOC-1434, which is “Verification approach with a level of extrinsic measures agreed to between the verification authority and the State”, moved to

be included in the modified INPR1.3.1 of URPR1.3 as an evaluation parameter, because the previous INPR2.2.2 is related to Extrinsic Measures.

For your references, the “**Variables**” of the Indicators described in the IAEA-TECDOC-1434 are shown as below.

Examples of intrinsic features are : Isotopic content of nuclear material; Chemical form of nuclear material, Radiation field from nuclear material; Heat generated by nuclear material; Spontaneous neutron generation rate from nuclear material; Complexity of, and time required for modifications necessary to use a civilian INS for a weapons production facility; Mass and bulk of nuclear material; Skills, expertise and knowledge required to divert or produce nuclear material and convert it to weapons useable form; Time required to divert or produce nuclear material and convert it to weapons useable form; Design features that limit access to nuclear material.

Examples of relevant commitments, obligations and policies include:

Safeguards agreements pursuant to the NPT; Export control policies; Relevant international conventions; Commercial, legal or institutional arrangements that control access to nuclear material and nuclear energy systems; Bilateral arrangements for supply and return of nuclear fuel; Bilateral agreements governing re-export of nuclear energy system components; Multi-national ownership, management or control of a nuclear energy system; Verification activities; State or regional systems for accounting and control; Safeguards approaches for the nuclear energy system, capable of detecting diversion or undeclared production; An effective international response mechanism for violations.

5.2 Description of new proposed Indicators

5.2.1 New Indicators of User Requirements of Basic Principle 1

A number of “Variables” were defined to evaluate the Indicators in IAEA-TECDOC-1434. However, we instead proposed the four Indicators and twelve evaluation parameters based on the previous variables, because Indicator can be evaluated by

aggregating the assessment results of the each evaluation parameter against the PR barrier. The modified Indicators of URPR1.1 are composed of four Indicators and each Indicator has several evaluation parameters as shown in Table 3.

Table 3. New proposed Indicators of URPR 1.1

BPPR1: PR features and measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that INSs will continue to be an unattractive means to acquire fissile material for a nuclear weapons programme.	
URPR1.1: The attractiveness of nuclear material and nuclear technology in an INS for a nuclear weapons programme should be low.	
INPR1.1.1: Material quality	EPPR1.1.1.1: Isotopic composition EPPR1.1.1.2: Material type EPPR1.1.1.3: Radiation field EPPR1.1.1.4: Heat generation EPPR1.1.1.5: Spontaneous neutron generation rate
INPR1.1.2: Material quantity	EPPR1.1.2.1: Mass of an item EPPR1.1.2.2: Number of items for Significant Quantity (SQ) EPPR1.1.2.3: Number of SQ (material stock or flow)
INPR1.1.3: Material form	EPPR1.1.3.1: Chemical/physical form
INPR1.1.4: Nuclear technology *Technology to provide weapon usable materials	EPPR1.1.4.1: Enrichment EPPR1.1.4.2: Extraction of fissile material EPPR1.1.4.3: Irradiation capability of target (reactor, accelerator)

The Indicators of URPR1.1 are explained as followings.

- **Indicator INPR1.1.1 (Material Quality)**

The material quality is evaluated in terms of the five evaluation parameters such as the isotopic composition, material type, radiation field, heat generation rate and the spontaneous neutron generation rate.

Isotope content

Highly enriched uranium or weapon grade plutonium is most attractive for a weapon application. For plutonium, ²³⁹Pu content (so-called plutonium quality)

is considered for the barrier evaluation in this study. For uranium, enrichment of ^{235}U or ^{233}U is examined for the same purpose.

Material type

Material type is a classification of nuclear material according to the contained element. For uranium, considering the degrees of enrichment, irradiated or unirradiated, the types which are defined at this study are as follows: Depleted Uranium (D), Natural Uranium (N), Low enriched Uranium (L), Direct Use of Un-irradiated Material (DUM), Direct Use of Irradiated Material (DIM).

Radiation field

The radiation field is a significant barrier to the accessibility because, if the radiation level is high, the shielding is required to access the nuclear material. The requirements of shielding material which is typically being heavy and cumbersome, and remote handling will necessitate the use of special lifting equipment, which makes such nuclear material less attractive. In this study, dose rate at 1 m from the surface of the nuclear material to be diverted is used as an indicator, and it can be evaluated by the radiation dose rate (Sv/hr).

Heat generation

The heat generation from the nuclear material complicates the facility operation and weapon design. Lower heat generation rate means a lower barrier compared to the higher heat generation rate. Since the heat generation rate depends on the ^{238}Pu concentration in the plutonium, the ^{238}Pu content is considered for the evaluation. Decay heat from fission products is mainly generated from ^{134}Cs , ^{137}Cs , ^{90}Sr and ^{137}mBa . Based on the calculation by ORIGEN code [4], the portion of decay heat generation from above three fission products is about 80% of total heat generation from all fission products

Spontaneous neutron generation rate

Spontaneous neutron can affect the design, the yield and the reliability of a nuclear explosive device. For plutonium, the spontaneous neutron production

depends on the concentration of ^{240}Pu and ^{242}Pu . Regarding the spontaneous neutron generation rate, the ratio of the plutonium isotope, $(^{240}\text{Pu} \text{ and } ^{242}\text{Pu})/\text{Pu}$, is used as an indicator in this study.

- **Indicator INPR1.1.2 (Material Quantity)**

Material quantity is evaluated in terms of evaluation parameters, which include mass of an item (kg), number of items for a SQ (Significant Quantity) and number of SQ during material stock or flow.

Mass of an item

It represents the evaluation parameter regarding how much mass one item has. If the weight of an item is heavy, its barrier is strong. Otherwise it is weak.

Number of items for SQ

It is the evaluation parameter to evaluate how many items are needed to divert one Significant Quantity of weapon useable materials.

Number of SQ

It is the evaluation parameter to evaluate how many Significant Quantity can be produced during the process in the viewpoint of the material stock and flow. It is evaluated by the number of SQ (number/year/facility).

- **Indicator INPR1.1.3 (Material Form)**

Material form refers to the extent and difficulty of the chemical process required to separate weapon-useable materials from their accompanying diluents and contaminants. If the compound is more contaminated, it is more difficult to convert the materials into metal form because it requires more time and processes to separate and treat the diverted materials. It is evaluated by the chemical or physical form of nuclear materials of the uranium, plutonium and thorium. The chemical forms of the nuclear materials can be the metal, oxide/solution, compound, spent fuel and the waste.

- **Indicator INPR1.1.4 (Nuclear Technology)**

The nuclear technology can be used for the production of the weapon usable materials. The evaluation parameters of the nuclear technology include the technologies for the “Enrichment process”, “Extraction process of the fissile material” and “Irradiation capability of a target material (e.g., reactor and/or accelerator).”.

Enrichment

Nuclear materials such as natural, depleted uranium(DU) and LEU require considerable enrichment steps to produce weapon-useable materials. Thus enrichment technology provides a barrier to proliferation resistant. Enrichment of natural uranium to HEU represents a higher barrier than that of LEU to HEU. Therefore, LEU has a slightly lower barrier to proliferation than natural or DU has.

Extraction of fissile material

The extent and difficulty of extraction of fissile material by chemical processing to separate the weapon-usable material would be chemical barrier. It is important to recognize that the radiological barrier in proliferation resistance aspects is more effective than chemical barrier to extract the fissile material.

Irradiation capability of target (reactor, accelerator)

Target materials to be transmuted in fast reactor or ADS(accelerator driven system) contain TRU-loaded fuel and long-lived fission products such as I-129, Tc-99. Fast reactor or ADS are preferable to transmute long-lived fission products, because the transmutation of fission products in LWR requires increasing the fissile enrichment to compensate insufficient excess neutrons. Irradiation of target materials in fast reactor or ADS renders long-lived fission products to short-lived fission products which are not considered as weapon-usable materials.

The Indicators of URPR1.2 are composed of five Indicators and each Indicator has several evaluation parameters as shown in Table 4.

Table 4. New proposed Indicators of URPR 1.2

BPPR1: PR features and measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that INSs will continue to be an unattractive means to acquire fissile material for a nuclear weapons programme.	
URPR1.2: The diversion of nuclear material should be reasonably difficult and detectable. *Diversion includes the use of an INS facility for introduction, production, processing of undeclared nuclear material.	
INPR1.2.1: Accountability	EPPR1.2.1.1: MUF/SQ EPPR1.2.1.2: NDA measurement capability by inspectors
INPR1.2.2: Applicability of C/S measures	EPPR1.2.2.1: Applicability of containment measures EPPR1.2.2.2: Applicability of surveillance measures EPPR1.2.2.3: Applicability of other monitoring systems
INPR1.2.3: Detectability of nuclear material	EPPR1.2.3.1: Possibility to identify nuclear material by NDA EPPR1.2.3.2: Hardness of radiation signature EPPR1.2.3.3: Need for passive/active mode
INPR1.2.4: Difficulty to modify the process	EPPR1.2.4.1: Extent of automation EPPR1.2.4.2: Availability of data for inspectors EPPR1.2.4.3: Authenticability of data to be provided for safeguards purpose EPPR1.2.4.4: Transparency of process EPPR1.2.4.5: Accessibility of material to inspectors
INPR1.2.5: Difficulty to modify facility design	EPPR1.2.5.1: Verifiability of facility design by inspectors

The Indicators of URPR1.2 are explained as followings.

- Indicator INPR1.2.1 (Accountability of Nuclear Material)

For the verification of the status of the material accounting data, the IAEA must be able to derive a statement of Material Unaccounted For (MUF) and a statistical Limit of Error for the MUF (LEMUF). The MUF is defined by the

IAEA as "the difference between the book inventory and the physical inventory". If a value represents one standard deviation of uncertainty in the MUF determination, then the amount of the diverted plutonium that could be detected with a 95% detection probability and a 5% false positive rate – the nominal safeguards goal – is 3.3 times the value. The MUF becomes a significant factor in bulk handling processes if the MUFs in these facilities could be beyond 1 SQ (8 kg of plutonium, 25 kg of ^{235}U over 20 wt% enrichment). Table 5 shows the definition of the SQ by the IAEA for various nuclear materials [5].

Table 5. Definition of the Significant Quantity by the IAEA

Material		SQ
Direct Use of Nuclear Material	Pu	8 kg Pu
	^{233}U	8 kg ^{233}U
	HEU ($^{235}\text{U} > 20\%$)	25 kg ^{235}U
Indirect Use of Nuclear Material	LEU ($^{235}\text{U} < 20\%$)	75 kg ^{235}U (or 10t natural U or 20t depleted U)
	Th	20t Th

And, Indicator INPR1.2.1 has two evaluation parameters such as MUF/SQ (MUF/Significant Quantity) and NDA (Non-destructive Assay) Measurement capability by inspectors.

MUF/SQ

MUF stands for Material Unaccount For and it is calculated for a material balance area (MBA) over a material balance period using the material balance equation, commonly written as:

$$\text{MUF} = (\text{PB} + \text{X} - \text{Y}) - \text{PE}$$

where,

PB is the beginning physical inventory,

X is the sum of increases to inventory,
Y is the sum of decreases from inventory,
PE is the ending physical inventory.

Because book inventory is the algebraic sum of PB, X and Y, MUF can be explained as the difference between the book inventory and the physical inventory. For item counting MBAs, MUF should be zero, and a non-zero MUF is an indication of a problem (e.g. accounting mistakes) which should be investigated. For bulk handling MBAs, a non-zero MUF is expected because of measurement uncertainty and the nature of processing. The operator's measurement uncertainties associated with each of the material balance areas are combined with the material quantities to determine the uncertainty of the material balance.

NDA Measurement Capability by Inspectors

Non-destructive assay (NDA) is a measurement of the nuclear material content, the element or isotopic concentration of an item without producing significant physical or chemical changes in the item. It is generally carried out by observing the radiometric emission or response from the item and by comparing that emission or response with a calibration based on essentially similar items whose contents have been determined through destructive analysis. There are two broad categories of NDA:

- (a) Passive analysis (assay), in which the measurement refers to spontaneous emissions of neutrons, gamma rays or the total decay energy;
- (b) Active analysis (assay), in which the measurement refers to a stimulated emission (e.g., neutron or photon induced fission).

- Indicator INPR1.2.2 (Applicability of C/S Measures)

This indicator considers three related evaluation parameters to monitor the nuclear material movement, such as applicability of the containment measures, applicability of the surveillance measures and applicability of other monitoring

systems. The use of C/S measures is aimed at verifying information on the movement of nuclear or other material, equipment and samples, or preservation of the integrity of safeguards relevant data. In many instances C/S measures cover the period when the inspector is absent, thus ensuring the continuity of knowledge for the IAEA and contributing to the cost effectiveness. Containment/Surveillance measures are applied in case of:

- (a) During flow and inventory verification to ensure that each item is verified without duplication and that the integrity of samples is preserved.
- (b) To confirm that there has been no change to the inventory, which was previously verified and thus reduce the need for remeasurement.
- (c) To ensure that IAEA equipment, working papers and supplies have not been tampered with.
- (d) If necessary, to isolate ('freeze') nuclear material that has not been verified until it can be measured.

The indication of an anomaly by C/S measures does not necessarily by itself indicate that material has been removed. The ultimate resolution of C/S anomalies is provided by nuclear material verification. If any C/S measures has been, or may have to be, compromised, the IAEA shall, unless agreed otherwise, be notified by the fastest means available. Examples of compromising might be seals which have been broken inadvertently or in an emergency, or seals of which the possibility of removal after advance notification to the IAEA has been agreed upon between the IAEA and the State.

The system of containment/surveillance is a combination of containment and/or surveillance measures. Each C/S system is designed to meet the purpose specified in the IAEA's safeguards approach. To increase reliability, a C/S system can include one or several C/S devices. C/S devices and containment may be used in such a way that a plausible diversion path is covered by at least one device (single C/S). For redundancy purpose, C/S devices may be backed up (duplicated) by a similar device. In a dual C/S system, each plausible diversion

path is covered by two C/S devices that are functionally independent and are not subject to a common tampering or failure mode (dual C/S), e.g., two different types of seal, or seals plus surveillance. Dual C/S is normally applied where the verification of nuclear material is difficult to perform in order to increase confidence in the C/S results and reduce the requirements for periodic reverification.

Applicability of the Containment Measures

It is a parameter which determines whether containment systems can apply to the system or not.

Applicability of the Surveillance Measures

It is a parameter which determines whether surveillance measures can apply to the system or not.

Applicability of other Monitoring Systems

It is a parameter which determines whether other monitoring systems such as Unattended monitoring, Remote monitoring, Core discharge monitor (CDM), Spent fuel bundle counter, Reactor power monitor and Radiation passage monitor, can apply to the system or not.

- Indicator INPR1.2.3 (Detectability of Nuclear Material)

This is evaluated by the nature of the detection system and the nuclear material to be detected. The evaluation parameters of the detectability include the possibility to identify nuclear material by NDA, the hardness of radiation signature and the need for passive/active mode

Possibility to Identify Nuclear Material by NDA

If nuclear material can be identified by NDA, this barrier is “Strong” against PR. Otherwise, it is “Weak” against PR.

Hardness of Radiation Signature

If the radiation signature from the nuclear materials is hard, this barrier is “Strong” against PR. Otherwise, it is “Weak” against PR.

Need for Passive/Active Mode

The passive system to detect the diversion of nuclear material has the measurement device for spontaneous emissions of neutrons or gamma rays or for the total decay energy. On the other hands, the Active system has the measurement device for a stimulated emission. (e.g., neutron or photon induced fission).

- Indicator INPR1.2.4 (Difficulty to Modify the Process)

Difficulty to modify the process depends on the complexity of the modification, cost for the process modification, safety implication of such modification, and the time required to perform the relevant modification. The evaluation parameters include five categories as followings.

Extent of Automation

The extent of automation for a process influences its degree for the modification. More process is automated, the stronger resistance against proliferation, because its access for diversion can not be easily made without detection. The batch operation is generally more difficult to automate the whole process, while compared to the continuous operation.

Availability of Data for Inspectors

If all the data on the process are readily available to the inspectors, the proper judgement regarding the modification or misuse of the process can be easily made, so it has strong proliferation resistance.

Authenticability of Data to be Provided for Safeguards Purpose

It is the measures providing the assurance that genuine information has originated from a known source (sensor) and has not been altered, removed or replaced. In the case of digital data, the use of certified authentication algorithms

contributes significantly to an adequate level of data authentication in unattended equipment systems.

Transparency of process

The transparency of process can make the modification of process difficult because it can be easier to be discovered by inspector. Generally, for well known process, they can induce the process history only with several main process data, but it is not easy to make up whole process for the complicated process or not-well established process.

Accessibility of Material to Inspectors for Verification

Concealment methods taken within a diversion strategy or a acquisition strategy for reducing of the probability of detection by IAEA safeguards activities can be used to divert a nuclear material in the fuel cycle facilities. Such actions may begin before the removal of material and may be continued over a considerable time. If creating obstacles against the access by IAEA inspectors so as to reduce the possibility of their detection can be easily installed, it will be low PR barrier.

- Indicator INPR1.2.5 (Difficulty to Modify Facility Design)

Verifiability of Facility Design by Inspectors

Difficulty to modify the fuel cycle facilities depends on the complexity of the modification, cost for the facility modification, safety implication of such a modification, and the time required to perform the relevant modification.

The Indicators of URPR1.3 are composed of two Indicators and each Indicator has several evaluation parameters as shown Table 6.

Table 6. New proposed Indicators of URPR 1.3

BPPR1: PR features and measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that INSs will continue to be an unattractive means to acquire fissile material for a nuclear weapons programme.	
URPR1.3: States' commitments, obligations and policies regarding non-proliferation and its implementation should be adequate to fulfill international standards.	
INPR1.3.1: States' commitments, obligations and policies regarding non-proliferation to fulfill international standards.	EPPR1.3.1.1: Safeguards agreements pursuant to the NPT EPPR1.3.1.2: Nuclear-weapons-free zone treaties EPPR1.3.1.3: Comprehensive IAEA safeguards agreements EPPR1.3.1.4: Additional protocols of IAEA agreements EPPR1.3.1.5: Export control policies of NM and nuclear technology EPPR1.3.1.6: Relevant international conventions EPPR1.3.1.7: State or regional systems for accounting and control EPPR1.3.1.8: Verification approach with a level of extrinsic measures agreed to between the verification authority and the State (it was come from old INPR2.2.2.)
INPR1.3.2: Facility/Enterprise undertakings to provide PR* *Appropriate wording should be found to formulate INPR1.3.2 with respect to providing "support or fulfill"	EPPR1.3.2.1: Multi-lateral ownership, management or control of a NES (Multi-lateral, Multi-National) EPPR1.3.2.2: International dependency with regard to fissile materials and nuclear technology EPPR1.3.2.3: Commercial, legal or institutional arrangements that control access to NM and NES

The Indicators of URPR1.3 are explained as followings.

- **Indicator INPR1.3.1 (States' commitments, obligations and policies regarding non-proliferation to fulfill international standards)**

It is necessary to evaluate the extrinsic measures related to State-specific information and contains eight evaluation parameters for the extrinsic measures

Safeguards Agreements Pursuant to the NPT

Non-Proliferation Treaty (NPT) was first introduced in 1968. There are two groups of States in the NPT, which have different duties and rights. Nuclear Weapon States, that already have nuclear weapons before the NPT was in effect, have the right to keep their nuclear weapons while the other States (Non-Nuclear Weapon States) can use nuclear energy only for peaceful purposes. This policy has played a great role in reducing the threat of nuclear weapons proliferation in the whole world. The fact if a nation has joined the NPT is used as an indicator. If the nation has not joined the NPT, it is regarded that the nation does not have any barrier against the NPT.

Nuclear-Weapons-Free Zone Treaties

There are several Nuclear Weapon-Free Zone Treaties worldwide such as the Treaty of Pelindaba of South Africa signed in 1995, Southeast Asia Nuclear Weapon-Free Zone Treaty in 1995, Tlateloco Treaty of South America in 1967 and the Rarotonga Treaty of the South Pacific region in 1986. The fact if a nation has joined the Nuclear Weapon-Free Zone Treaty is used as an indicator.

Comprehensive IAEA Safeguards Agreements

Comprehensive safeguards agreement (CSA) is an agreement that applies safeguards to all nuclear material in all nuclear activities in a State. CSAs can be grouped as follows [6].

- (a) A safeguards agreement pursuant to the NPT, concluded between the IAEA and a non-nuclear-weapon State party as required by Article III.1 of the NPT. Such a safeguards agreement is concluded on the basis of [INFCIRC/153]. The agreement is comprehensive as it provides for the IAEA's right and obligation to ensure that safeguards are applied "on all source or special fissionable material in all peaceful nuclear activities within the territory of the State, under its jurisdiction, or carried out

under its control anywhere...” [INFCIRC/153, para. 2]. The scope of a CSA is not limited to nuclear material actually declared by a State, but includes any nuclear material that should have been declared to the IAEA. There may be non-peaceful uses of nuclear material which would not be proscribed under the NPT and to which safeguards would not apply during the period of such use (e.g., nuclear propulsion of submarines or other warships).

- (b) A safeguards agreement pursuant to the Tlatelolco Treaty or some other nuclear weapon-free-zone (NWFZ) treaty. The majority of States party to such treaties are also party to the NPT and each has concluded a single safeguards agreement which refers expressly to both the NPT and the relevant NWFZ treaty or which has subsequently been confirmed as meeting the requirements of both treaties.
- (c) A safeguards agreement, such as the *sui generis* agreement between Albania and the IAEA, and the quadripartite safeguards agreement between Argentina, Brazil, ABACC and the IAEA.

Additional Protocols of IAEA Agreements

Additional protocol is a protocol additional to a safeguards agreement (or agreements) concluded between the IAEA and a State, or group of States, following the provisions of the Model Additional Protocol INFCIRC/540. A comprehensive safeguards agreement, together with an additional protocol, contains all of the measures included in INFCIRC/540. In the case of an INFCIRC/66-type safeguards agreement or of a voluntary offer agreement, an additional protocol includes only those measures from INFCIRC/540 that have been agreed to by the State concerned. Under Article 1 of INFCIRC/540, the provisions of the additional protocol prevail in the case of conflict between the provisions of the safeguards agreement and those of the additional protocol.

Export Control Policies of NM and Nuclear Technology

There are several committees on export control such as the Zangger committee in 1995, Nuclear Supplier Group (NSG) in 1995 and the Wassenaar arrangement on export controls for conventional arms and dual-use goods and technology in 1996. The indicator will be determined based on the fact that a nation has export control policies and has joined one of the above-mentioned committees.

Relevant International Conventions

In this part, proliferation relevant international conventions such as CTBT (Comprehensive Test Ban Treaty) are considered.

State or Regional Systems for Accounting and Control

In this part, state safeguards system or regional safeguards system such as EURATOM are considered.

Verification Approach with a Level of Extrinsic Measures Agreed to between the Verification Authority and the State

Robustness of verification approach agreed to between the verification authority and the State will be considered.

- **Indicator INPR1.3.2 (Facility/Enterprise Undertakings to Provide PR)**

This is evaluated through the review of facility/enterprise undertakings to provide PR and contain three evaluation parameters for the extrinsic measures.

Multi-national Ownership Management or Control of an NES (Multi-lateral/Multi-national)

International ownership of nuclear material can definitely reduce proliferation risk. So there has issued several ideas related to the international ownership such as International Nuclear Fuel Storage and International Plutonium Management Concept. However, all of them are not yet substantiated.

International Dependency with regard to Fissile Materials and Nuclear Technology

The degree of the International Dependency with regard to Fissile Materials and Nuclear Technology will be considered.

Commercial, Legal or Institutional Arrangements that Control Access to Nuclear Material and Nuclear Energy Systems.

This can include the use of multi-national fuel cycle facilities, and arrangements for spent fuel take-back. (Commercial, legal or institutional arrangements that control access to NM and NES, Relevant international conventions, Multi-lateral ownership, management or control of a NES)

5.2.2 New Indicators of User Requirements of Basic Principle 2

The indicators of URPR2.1 are same as the previous one, and are composed of two indicators and each indicator has one evaluation parameter as shown in Table 7.

Table 7. New proposed Indicators of URPR 2.1

BPPR2: Both intrinsic features and extrinsic measures are essential, and neither shall be considered sufficient by itself.	
URPR2.1: Innovative nuclear energy systems should incorporate multiple PR features and measures.	
INPR2.1.1: The extent by which the INS is covered by multiple intrinsic features	EPPR2.1.1.1: “No. of plausible acquisition paths covered by multiple PR features and measures” and “No. of plausible acquisition paths”
INPR2.1.2: robustness of barriers covering an acquisition path	EPPR2.1.1.2: Extent of robustness of barriers

The Indicators of URPR2.1 are explained as followings;

- **Indicator INPR2.1.1 (Extent by which the INS is covered by multiple intrinsic features)**
 - The “Extent” is the fraction of plausible acquisition paths. It is understood that each acquisition path is covered by appropriate verification measures.

The evaluation parameter of this indicator is “Ratio of number of plausible acquisition paths covered by multiple barriers to the number of all plausible acquisition paths”.

- It is not explained in details how to evaluate INPR2.1.1 in IAEA-TECDOC-1434. Hence, this study proposed the steps for the evaluation of this Indicator as below.
 - ✓ Step 1: Define plausible acquisition paths by acquisition path analysis.
 - ✓ Step 2: Determine the plausible acquisition paths which are covered by PR features and measures, which are evaluated as at least “Moderate” and better.
 - ✓ Step 3: Calculate the ratio of “Number of plausible acquisition paths covered by multiple PR features and measures” to “Number of plausible acquisition paths”.

- Indicator INPR2.1.2 (Robustness of Barriers Covering an Acquisition Path)

The robustness indicates the strength of the barrier against the destruction, or to divert the nuclear material along the acquisition path. The evaluation parameter is “Extent of robustness of barriers”.

- This Indicator is important, but not explicitly explained during the development of the evaluation methodology. In the present study, the evaluation steps are setup as below.
 - ✓ “Robustness” can be described by the Evaluation Parameters and Indicators determined by BPPR1. (which are needed to be aggregated)
 - ✓ Step 1: Determine the robustness of each barrier for each plausible acquisition path separately.
 - ✓ Step 2: Aggregate the results.
- Step 1 will deliver the “raw data” for further aggregation. Aggregation of more than one barrier for an acquisition path and its comparison with single barriers will be subject to further study. Problems, for example, to be solved are:

- ✓ Is a single strong barrier more robust than two moderate barriers?
- ✓ Should one introduce weighing factors for intrinsic features and extrinsic measures?

The indicator of URPR2.2 is related to the cost to incorporate those intrinsic features and extrinsic measures, which is composed of one Indicator as shown in Table 8.

Table 8. New proposed Indicators of URPR 2.2

BPPR2: Both intrinsic features and extrinsic measures are essential, and neither shall be considered sufficient by itself.	
URPR2.2: The combination of intrinsic features and extrinsic measures, compatible with other design considerations, should be optimized (in the design/engineering phase) to provide cost-efficient proliferation resistance.	
INPR2.2.1: Cost to incorporate those intrinsic features and extrinsic measures, which are required to provide PR	EPPR2.2.1.1: Sum of costs

The indicator INPR2.2.1 is explained as followings;

- Indicator INPR2.2.1 (Cost to incorporate those intrinsic features and extrinsic measures, which are required to provide proliferation resistance)

The costs for the installation of the new intrinsic features and extrinsic measures or the modification of the existing intrinsic features and extrinsic measures are considered. The evaluation parameter of INPR2.2.1.1 is the amount of the sum of costs.

- This study suggested the steps to evaluate the cost effectiveness as below.
 - ✓ Step 1: Define the basic design characteristics of a nuclear installation ensuring economical and safe operation according to State's requirements.
 - ✓ Step 2: Define additional intrinsic features resulting from the acquisition path analyses in order to improve PR and to support the implementation of safeguards.

- ✓ Step 3: Estimate the costs for the additional design features and the costs for the implementation of safeguards. (based on current experience)
- ✓ Step 4: Determine the combination of intrinsic features and extrinsic measures for the minimum of total PR costs.
- ✓ Step 5: Compare the total PR costs of a proposed design of an INS with the optimal solution determined by Step 4.
- ✓ Presentation of the results will be subject to further study.

5.3 Scales for evaluation parameters

Some barriers can be quantified but the other barriers, such as extrinsic measures or safeguardability, may be expressed only in a logical value such as “Yes” or “No”. The present case study suggests five stage scale such as VW(Very Weak), W(Weak), M(Moderate), S(Strong) and VS(Very Strong) regarding the quantifiable evaluation parameters. For logical scale, U(Unacceptable) and A(Acceptable) for extrinsic measures and W(Weak) and S(Strong) for some intrinsic features related to safeguardability are suggested. The evaluation scale for each Indicator is shown in Tables 9 to 13. Most quantified scales of evaluation parameters in tables are referenced in [7].

Table 9. Evaluation scales of URPR 1.1

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Material quality	Isotopic composition	$^{239}\text{Pu}/\text{Pu}$ (wt%)	> 93	80~93	70~80	60~70	< 60
		$^{235}\text{U}/\text{U}$ (wt%)	> 90	50~90	20~50	5~20	< 5
		$^{232}\text{U}_{\text{contam.}}$ for ^{233}U (ppm)	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	$^{238}\text{Pu}/\text{Pu}$ (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	($^{240}\text{Pu} + ^{242}\text{Pu}$)/Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

Notes:

- For material type,
D: Depleted Uranium, N: Natural Uranium, L: Low enriched U, DUM: Direct Use of Un-irradiated Material, DIM: Direct Use of Irradiated Material.
- Each evaluation scale in the table includes a lower value of the range.

Table 10. Evaluation scales of URPR 1.2

Indicators	Evaluation Parameter		Evaluation scale				
			W		S		
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No		Yes		
	Applicability of surveillance measures		No		Yes		
	Applicability of other monitoring systems		No		Yes		
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No		Yes		
	Hardness of radiation signature		No reliable signature		Reliable signature		
	Need for passive/active mode		Active		Passive		
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for safeguards purpose		No		Yes		
	Transparency of process		No		Yes		
	Accessibility of material to inspectors for verification		No		Yes		
Difficulty to modify facility design	Verifiability of facility design by inspectors		No		Yes		

Table 11. Evaluation scales of URPR 1.3

Indicators	Evaluation Parameter	Evaluation scale	
		U	A
States' commitments, obligations and policies regarding non-proliferation to fulfill international standards.	Safeguards agreements pursuant to the NPT	No	Yes
	Nuclear-weapons-free zone treaties	No	Yes
	Comprehensive IAEA Safeguards agreements	No	Yes
	Additional protocols of IAEA agreements	No	Yes
	Export control policies of NM and nuclear technology	No	Yes
	Relevant international conventions	No	Yes
	State or regional systems for accounting and control	No	Yes
	Verification approach with a level of extrinsic measures agreed to between the verification authority and the State	No	Yes
Facility/Enterprise undertakings to provide PR *Appropriate wording should be found to formulate INPR1.3.2 with respect to providing "support or fulfill"	Multi-lateral ownership, management or control of a NES (Multi-lateral/Multi-National)	No	Yes
	International dependency with regard to fissile materials and nuclear technology	No	Yes
	Commercial, legal or institutional arrangements that control access to NM and NES	No	Yes

Table 12. Evaluation scales of URPR 2.1

Indicators	Evaluation Parameter	Evaluation scale				
		VW	W	M	S	VS
INPR2.1.1: The extent by which the INS is covered by multiple intrinsic features	"No. of plausible acquisition paths covered by multiple PR features and measures" and "No. of plausible acquisition paths"	< 0.2	0.2 ~ 0.4	0.4 ~ 0.6	0.6 ~ 0.8	> 0.8
INPR2.1.2: Robustness of barriers covering an acquisition path	Extent of robustness of barriers	Very little	Little	Medium	Great	Very great

Table 13. Evaluation scales of URPR 2.2

Indicators	Evaluation Parameter	Evaluation scale				
		VW	W	M	S	VS
INPR2.2.1: Cost to incorporate those intrinsic features and extrinsic measures, which are required to provide PR * TBD for Scales and Unit	Sum of costs					Minimum

6. Application of the Modified Methodology to the Whole DUPIC Fuel Cycle

6.1 General description

The whole DUPIC fuel cycle in Korean situation is defined as encompassing from supply of LEU from foreign country, PWR fuel fabrication, PWR spent fuel transportation, DUPIC fuel fabrication, to DUPIC spent fuel disposal. The system elements of the whole DUPIC fuel cycle can be classified into 7 system elements of PWR fuel cycle part and 8 system elements of DUPIC fuel cycle part as followings.

PWR fuel cycle part

- Step P1: Supply of LEU feed uranium from a foreign country for the PWR fuel fabrication
- Step P2: Transportation of LEU material to PWR fuel fabrication Facility
- Step P3: PWR fuel fabrication facility
- Step P4: Transportation of PWR fuel to PWR plant
- Step P5: PWR plant
- Step P6: Transportation of PWR spent fuel to interim storage facility
- Step P7: Interim storage of PWR spent fuel

DUPIC fuel cycle part

- Step D1: Transportation of PWR spent fuel to DUPIC fuel fabrication facility
- Step D2: DUPIC fuel fabrication plant
- Step D3: Transportation of DUPIC fuel to CANDU plant
- Step D4: CANDU plant
- Step D5: Transportation of DUPIC spent fuel to interim storage facility
- Step D6: Interim storage of DUPIC spent fuel
- Step D7: Transportation of DUPIC spent fuel to permanent disposal facility
- Step D8: Permanent disposal of DUPIC spent fuel

Seven system elements among the above system elements of the whole fuel cycle of DUPIC are considered at the present study. That is, the yellow colored box as shown in Fig. 9 are evaluated with new modified PR methodology proposed by this case study.

The reasons for choosing seven system elements for this case study are: (1) the system elements of mining and enrichment would not occur in Korea and this stages are conducted outside of Korea. Therefore, it is extremely resistant to proliferation by the DUPIC State. Hence, these system elements are not included for the proliferation resistance evaluation. (2) One system element of “Transportation of PWR spent fuel to DUPIC fabrication plant” is considered because the proliferation resistance characteristics of other system elements regarding “Transportation” may be very similar to “Transportation of PWR spent fuel to DUPIC fabrication plant” in view points of physical protection or safeguards even if its material properties can be slightly different each other. (3) As the CANDU plant is evaluated in terms of proliferation resistance barriers, the PWR plant can be evaluated in same manner of CANDU plant because the Korean Extended Case Study is to assess the adequacy of the new modified INPRO Methodology and to aim at recommendations for improvement of the INPRO methodology. (4) Also, the assessment of the proliferation resistance of “Interim Storage of PWR spent fuel” is not included at this study in the same rationale because the assessment of proliferation resistance of “Interim Storage of DUPIC spent fuel” is considered.

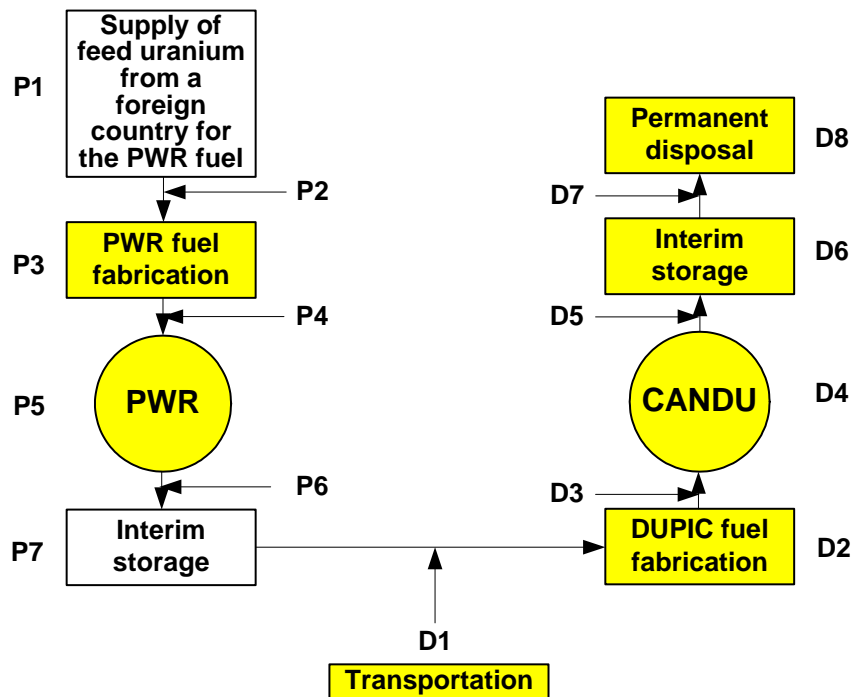


Figure 9. Selected system elements for PR assessment of the whole DUPIC fuel cycle

6.2 Material characteristics of the whole DUPIC fuel cycle

In order to evaluate the PR characteristics of DUPIC fuel cycle, the material flow was calculated based on the assumption of 10 GWe-year for the scale of the whole DUPIC fuel cycle as shown in Fig. 10.

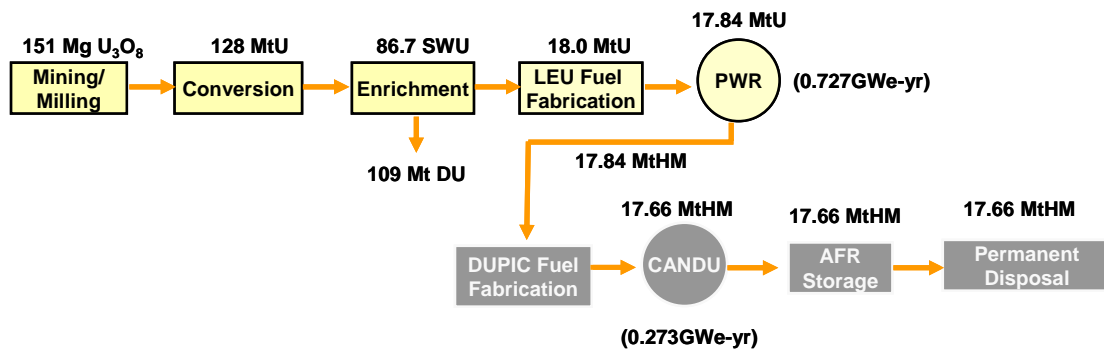


Figure 10. Material flow of the whole DUPIC fuel cycle

Equilibrium reactor ratio of DUPIC plant to PWR plant is about two and hence, the portion of PWR and CANDU electricity power generation is 73% and 27%,

respectively. Based on this assumption, the plutonium isotopes and radiation fields in the DUPIC fuel cycle are determined as Table 14 and Table 15, respectively.

Table 14. Pu isotope composition in various spent fuels

Isotopes	PWR SF		Fresh DUPIC Fuel		DUPIC SF	
	g/MtHM	wt% of Pu	g/MtHM	wt% of Pu	g/MtHM	wt% of Pu
PU238	1.54E+02	1.7	1.54E+02	1.7	3.88E+02	4.9
PU239	5.33E+03	59.9	5.33E+03	59.9	3.16E+03	39.7
PU240	2.20E+03	24.8	2.20E+03	24.8	2.79E+03	35.1
PU241	7.52E+02	8.4	7.52E+02	8.4	5.24E+02	6.6
PU242	4.57E+02	5.1	4.57E+02	5.1	1.10E+03	13.8

Table 15. Dose rates of various nuclear fuels

Items		Dose rate (Sv/hr) for diversion of one assembly or one bundle	Total dose rate (Sv/hr) for 1000kgHM diversion	Dose rate (Sv/hr) for diversion of 1 SQ (8 kg Pu)
PWR SF	35 GWD/MtU, 10 yrs cooling	10.37	23.56	21.21
Fresh DUPIC Fuel	PWR SF (35 GWD/MtU, 10 yrs cooling)	0.15	7.97	7.17
DUPIC SF	15 GWD/MtU, 10 yrs cooling	0.61	32.16	32.32
CANDU SF	7.5 GWD/MtU, 10 yrs cooling	0.22	11.51	22.84

6.3 PR evaluation of each DUPIC fuel cycle system element

6.3.1 PR evaluation of Extrinsic Measures (URPR1.3 of BPPR1)

The assessment of proliferation resistance of the extrinsic measures, URPR1.3 of BPPR1, is not dependent on the system elements but on the States. Hence, extrinsic measures are evaluated at first considering the current Korean situation.

Safeguards Agreements Pursuant to the NPT

Korea joined the NPT as a non-nuclear weapon State in 1975 and supported the extension of the NPT for an indefinite duration without any condition in 1995.

Nuclear-Weapons-Free Zone Treaties

Regarding the nuclear weapon-free zone treaties, there is a similar agreement around Korean peninsula. For example, North and South Korea signed the joint declaration on the denuclearization of the Korean peninsula. And the joint declaration officially entered into force on February 19, 1992 and still remains valid. This was confirmed and reconfirmed at the June 2001 summit in Pyongyang. For the CTBT, it is open for the signature of each country based on the U.N. resolution. Korea signed the treaty in 1996.

Comprehensive IAEA Safeguards Agreements

Korea signed the INFCIRC/153 type agreement, “Agreement between Korea and IAEA for the application of safeguards in connection with the treaty on the non-proliferation of nuclear weapons”, in 1975.

Additional Protocols of IAEA Agreements

Korea signed and ratified the Additional Protocol to the Agreement(s) between State(s) and IAEA for the application of safeguards (INFCIRC/540) in 2004.

Export Control Policies of NM and Nuclear Technology

Korea is strongly against nuclear weapons proliferation and is in favor of exercising necessary control and international supervision over nuclear material

transfer so as to prevent the proliferation of nuclear weapons and related technology. From this position, Korea joined the Zangger Committee in 1995, Nuclear Supplier Group (NSG) in 1995 and the Wassenaar Arrangement on Export Controls for Conventional Arms and Dual-Use Goods and Technology in 1996.

Relevant International Conventions

The IAEA has no authority to take coercive measures to stop or reverse nuclear proliferation. Therefore it reports to the U.N. Security Council, by which the U.N. Security Council may take forceful measures against proliferation under the U.N. Charter VII.

State or Regional Systems for Accounting and Control

Concerning State or regional systems for accounting and control, the Korean Government enacted a nuclear law on national safeguards activities and established a mandatory body; that is, a Technology Center for Nuclear Control (TCNC) was founded in 1997. Since then, a national inspection was performed for all the facilities with nuclear material in Korea. But there is no regional system for accounting and control around Korea even though there is the ASIATOM concept which is similar to the EURATOM concept.

Verification approach with a level of extrinsic measures agreed to between the verification authority and the State

According to bilateral safeguards agreement between Korea and IAEA, the Design Information Questionnaire (DIQ) of the nuclear facilities in Korea are reported to IAEA from the beginning stage of the construction and then Design Information Verification (DIV) is performed by the IAEA. The safeguard approach as well as design information are included in the DIQ. And then IAEA designs appropriate verification approach including containment and surveillance with the DIQ. Therefore it would be said that the verification approach with a level of extrinsic measures agreed to between the IAEA and Korea is good or robustness.

Multi-lateral ownership, management or control of a NES (Multi-lateral/Multi-National)

Concerning multi-lateral ownership, management or control of a NES such as bilateral agreements for the supply and return of the nuclear fuel, Korea has imported nuclear material mainly from Australia, Canada and the U.S.A. By the bilateral agreements, suppliers have the right to ask for the return of the nuclear material if Korea uses non-peacefully the transferred nuclear material. Regarding the bilateral agreements governing the re-export of nuclear energy system components, Korea has entered into nuclear cooperation agreements with many countries including Canada, France, Japan and the U.S.A. The re-export of components of the nuclear energy system have been controlled through these agreements.

International dependency with regard to fissile materials and nuclear technology

Korea depends heavily on nuclear power for its electricity generation with 20 nuclear power units in operation, sharing 40 percent of the total production of electricity. Being poorly endowed with uranium reserves, whole uranium is imported from foreign countries. Regarding nuclear technology, Korea has increased especially for nuclear power plant technology, and it is known that its localization ratio for nuclear power plant technology has reached to almost 95%. On the other hand, technology on nuclear fuel cycle is still depending on foreign countries. On the whole, it would be said that international dependency of Korea with regard to fissile materials and nuclear technology is “large”.

Commercial, legal or institutional arrangements that control access to NM and NES

International ownership of the nuclear material can definitely reduce the proliferation risk. So there have been several ideas related to the international ownership such as the international nuclear fuel storage and international plutonium management concepts. However all of them have not yet been substantiated.

Table 16. Evaluation of Extrinsic Measures of URPR 1.3

Indicators	Evaluation Parameter	Evaluation scale	
		U	A
States' commitments, obligations and policies regarding non-proliferation to fulfill international standards.	Safeguards agreements pursuant to the NPT	No	Yes
	Nuclear-weapons-free zone treaties	No	Yes
	Comprehensive IAEA Safeguards agreements	No	Yes
	Additional protocols of IAEA agreements	No	Yes
	Export control policies of NM and nuclear technology	No	Yes
	Relevant international conventions	No	Yes
	State or regional systems for accounting and control	No	Yes
	Verification approach with a level of extrinsic measures agreed to between the verification authority and the State	Bad	Good
Facility/Enterprise undertakings to provide PR *Appropriate wording should be found to formulate INPR1.3.2 with respect to providing “support or fulfill”	Multi-lateral ownership, management or control of a NES (Multi-lateral/Multi-National)	No	Yes
	International dependency with regard to fissile materials and nuclear technology	Small	Large
	Commercial, legal or institutional arrangements that control access to NM and NES	No	Yes

6.3.2 PR Evaluation of “PWR fuel fabrication” (Step P3)

The system characteristics of PWR fuel fabrication for PWR plant are as followings.

The input material to PWR fuel fabrication facility is low enriched UO_2 powder and the maximum enrichment handled by this facility is assumed as 3.5 wt%. The facility capacity is 180 tU which is based on 10 GWe-year in PWR and CANDU with DUPIC fuel.

The PWR fuel fabrication process consists of six major manufacturing processes as shown in Fig. 11 and the characteristics of each process are as followings.

- UO_2 powder processing

- Receiving of UO_2 powder

UO_2 powder in transport container enters into a buffer storage located at facility. The UO_2 powder produced at conversion plant in a same lot is counted as a same batch for accounting purposes. For weighing and

sampling of UO_2 powder, one sample for each batch in random sampling basis is taken and sent to the laboratory for measuring tare weight in transport container.

- Powder treatment

UO_2 powder is transported pneumatically to blender and is blended to adjust powder size in a cell, if necessary. The blended powders are fed into the pellet fabrication units. The amount of batch process is limited by safety regulation.

- Fuel pellet fabrication

- Pelletizing of UO_2 powder

The UO_2 powders are first compacted mechanically to produce green pellets of about 60% density in enclosed and ventilated area by the automated pelletizing equipment. The green pellets are put on a boat and placed in temporary storage area.

- Fabrication of sintered pellet

The green pellets are sintered in hydrogen atmosphere at $1,750\text{ }^\circ\text{C}$ to a density of 95%. The sintered pellets are then ground to exact dimensions and inspected automatically. The pellets ground by wet grinder are dried in elevated temperature furnace. Visual examination is carried out to eliminate pellets with cracks, and good quality pellets are stored in trays.

- Fuel assembly

- Fuel rod fabrication

Columns of sintered pellets are weighed and loaded into fuel cladding tubes. The fuel pellets are loaded automatically into Zircaloy tubes from corrugated tray using vibratory feed. The load tubes are seal-welded to make fuel rods. Finished fuel rods are examined according to quality control guidelines. The examination includes non-destructive assay of fuel enrichment for each fuel rod.

- Fuel assembly fabrication

Finished fuel rods are transferred to the assembling area. Then the rods are assembled to make fuel assembly and they are temporarily stored on hangers.

- Recovery of scrap

- Scrap material consists mainly of grinder slag, defective sintered pellets, etc.
- This recoverable scrap is converted into U_3O_8 and processed through scrap recovery procedure to produce green powder, and it will be recycled.
- The scrap recycle rate at the PWR fuel fabrication facility is assumed as about 7% of throughput.

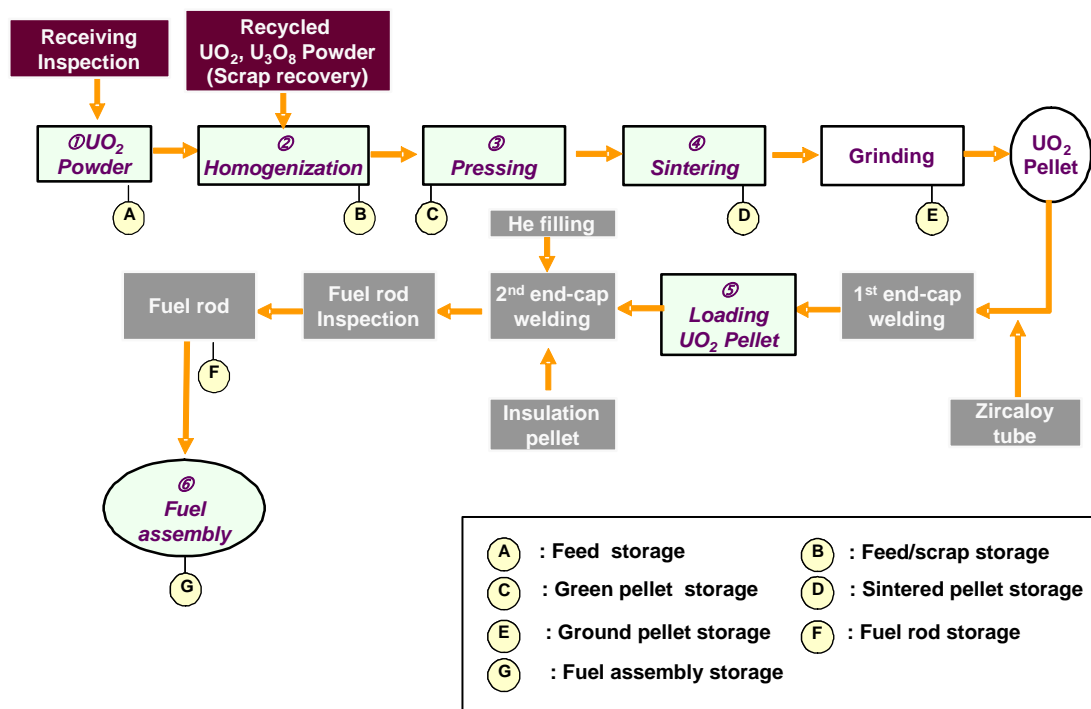


Figure 11. Manufacturing processes of PWR fuel

The proliferation resistance characteristics of PWR fuel fabrication are evaluated as followings.

The PWR fuel fabrication facility has a nature of bulk handling process. Therefore, MUF for the measurement of nuclear material should be considered. The enrichment in LEU fuel is assumed as about 3.5 wt%, and plutonium content is not existed. The Manufacturing Automation Process (MAP) for automated production of LEU fuel operates in a continuous flow mode. A single operator in central control facility can monitor the entire production with the aid of signals from microprocessors.

Inspection of the configuration of pellets and their enrichment are performed by gamma scanning and various measurement systems for mass determination of nuclear materials for each process, considering the chemical and physical characteristics of nuclear materials.

The rationale of the PR assessment results of Indicators for User Requirement 1.1 for PWR fuel fabrication are described below.

- User Requirement 1.1

- INPR1.1.1: Material quality
 - ✓ Material type: 3.5 % low enriched uranium
 - ✓ Isotopic composition: $^{239}\text{Pu}/\text{Pu} = 0 \text{ wt}\%$
 - ✓ Radiation field: Dose rate of a PWR fuel assembly is very low(<0.1 mSv/hr)
 - ✓ Heat generation: $^{238}\text{Pu}/\text{Pu} = 0 \text{ wt}\%$
 - ✓ Spontaneous neutron generation rate: $(^{240}\text{Pu}+^{242}\text{Pu})/\text{Pu} = 0 \text{ wt}\%$
- INPR1.1.2: Material quantity
 - ✓ Mass of an item: ~ 450 kg
 - ✓ No. of items to get one Significant quantity (SQ): no Pu
 - ✓ No. of Significant Quantities which can be attained through LEU fuel fabrication process (no Pu and ^{233}U)
- INPR1.1.3: Material form
 - ✓ Uranium oxides: no change of chemical form
- INPR1.1.4: Nuclear technology

- ✓ The PWR fuel fabrication technology employs a typical powder/pelletizing process and there is no chemical process involved. It is difficult to modify the facility and processes for enrichment.

Table 17. Evaluation of URPR1.1 of Step P3

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Material quality	Isotopic composition	²³⁹ Pu/Pu (wt%)	> 93	80~93	70~80	60~70	< 60
		²³⁵ U/U (wt%)	> 90	50~90	20~50	5~20	< 5
		²³² U _{contam.} for ²³³ U (ppm)	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	²³⁸ Pu/Pu (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	(²⁴⁰ Pu+ ²⁴² Pu)/Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

- User Requirement 1.2

□ Accountability

- ✓ MUF: ~ 180,000kg/year x 0.0005(measurement error) x 1/3 (material balance period, 4months) = 30kgLEU
- ✓ A Near Real Time Accounting System (NRTA) for fissile accountability system shall be used in the plant. The NRTA system is integrated with individual nuclear material measurement system.

- ✓ The item accounting for both PWR fuel incoming and outgoing shall be measured and recorded. The weighing and NDA systems for the bulk accounting in the process would be applied.
- Applicability of C/S measures
 - ✓ The containment and surveillance systems can be easily installed in the facility.
 - ✓ Applicability of other monitoring systems
 - ✓ Feed material measurement: balance or weighing system
 - ✓ Process monitoring: Unattended continuous process monitoring system
- Detectability of nuclear material

Various measurement methods for detection of nuclear material for each of the flow in PWR fuel fabrication facility. Inventory key measurement points are established in consideration of physical and chemical characteristics of nuclear materials. Mass measurement and analytical chemical measurement are used for the measurement of the percent of ^{235}U .
- Difficulty to modify the process

The LEU fuel fabrication process handles low enriched UO_2 powder and pellet with determined content. Therefore, it is very difficult to modify the process and facility for obtaining highly enriched ^{235}U from LEU material.
- Difficulty to modify facility design
 - ✓ Verifiability of facility design by inspectors: The PWR fuels are fabricated in limited open area. Access to the nuclear materials is relatively easy because of the low radiation field.

Table 18. Evaluation of URPR1.2 of Step P3

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No			Yes	
	Applicability of surveillance measures		No			Yes	
	Applicability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Hardness of radiation signature		No reliable signature			Reliable signature	
	Need for passive/active mode		Active			Passive	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for safeguards purpose		No			Yes	
	Transparency of process		No			Yes	
	Accessibility of material to inspectors for verification		No			Yes	
Difficulty to modify facility design	Verifiability of facility design by inspectors		No			Yes	

6.3.3 PR evaluation of “PWR plant” (Step P5)

The PR characteristics of PWR plant related to fuel are explained as follows.

The facilities related to fuel in PWR plant are comprised of new fuel storage racks, spent fuel storage racks and reactor. The new fuel storage racks are used for the dry storage of new fuel assemblies required for refuelling the reactor. The racks are located in the new fuel storage area inside the fuel building as shown in Fig. 12 and are designed to provide vertical storage for new fuel assemblies. The total capacity of the new fuel storage racks is generally ~50 assemblies corresponding to about 1/3 of full

core. The spent fuel storage racks are used for temporary underwater storage of spent fuel assemblies following discharge from the core and prior to shipment for the next treatment such as reprocessing. The racks are located in the spent fuel storage area inside the fuel building as shown in Fig. 12 and are designed to include storage for about 500 fuel assemblies corresponding to more than three full cores. The reactor is located inside the containment as shown Fig. 12 and are designed to charge 121 fuel assemblies. The total capacity of the fuel storage is dependent on the plant design.

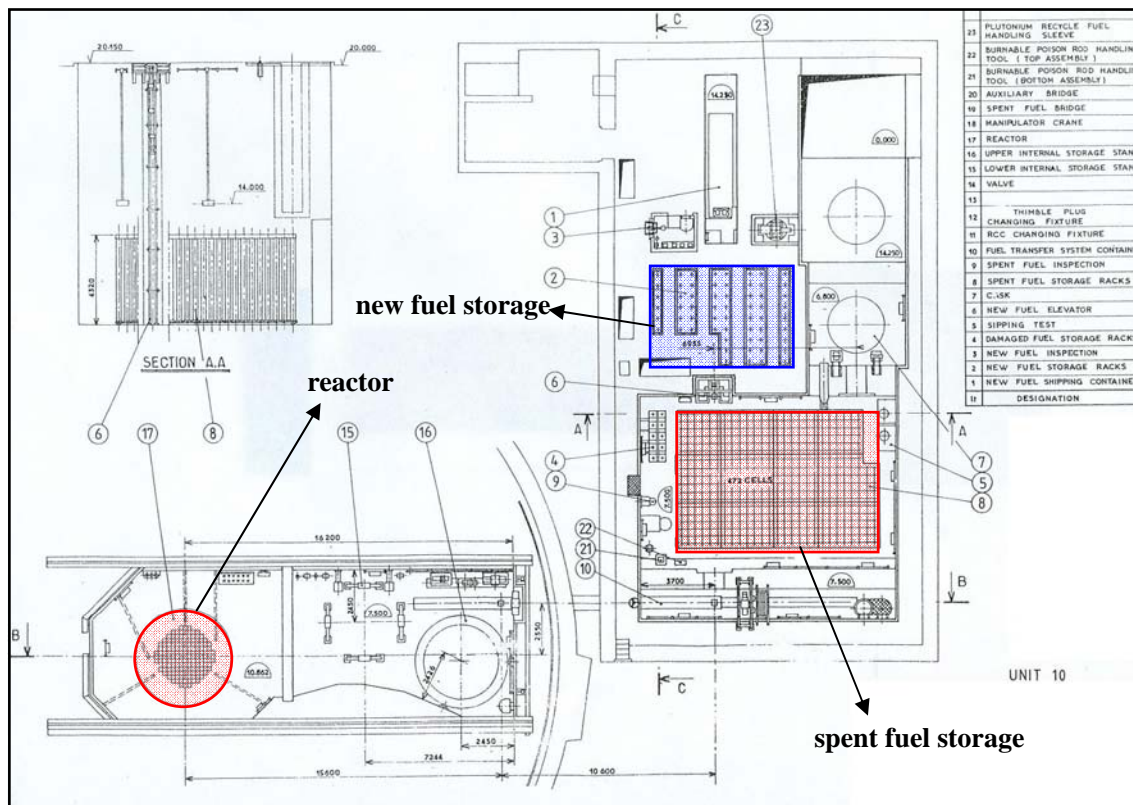


Figure 12 The layout of the reactor and fuel building in a PWR plant

The reactor is continuously kept full of water during reactor operation and shutdown for refuelling, and is inaccessible to the personnel. The fuel building comprising the new fuel storage area and the spent fuel storage area is accessible to the personnel. But spent fuel assemblies should be always stored and handled in a sufficient depth of water to ensure adequate biological protection of personnel against radioactivity.

The PWR fuel assemblies are remotely loaded into the core and discharged from the core by using the fuel handling system such as manipulator crane, spent fuel pit bridge

crane, fuel transfer system, etc. The integrity of the fuel is monitored by visual inspection when the reactor vessel is opened for refuelling or inspection.

During the normal and abnormal operations, there is no way that the fuel assemblies are repositioned without using the fuel handling system. Passages to the reactor building are the equipment lock and spent fuel transfer canal.

If new PWR fuel is arrived at the PWR plant, fuel assemblies are counted and stored in the new fuel storage racks inside the fuel building. The new PWR fuel assemblies go through visual inspection before loading to a reactor. A fuel manipulator crane moves the new PWR fuel assembly from the new fuel rack to the spent fuel storage pit. Once the fuel assembly is transferred to the spent fuel storage pit, the spent fuel pit bridge crane places the fuel assembly in the fuel rack. The fuel assemblies are loaded into the reactor core following the refuelling scheme. The average fuel residence time in the reactor is 1000 days. The fissile content of the PWR fuel is about 3.5 wt% when the fuel loaded, while it is 1.5 wt% when discharged.

The integrity of the fuel during the normal operation is monitored by the radiation level of the coolant. As the new fuel assemblies are loaded, the burnt PWR fuel assemblies are discharged from the core and transferred to the spent fuel pit. The spent fuels are inspected for the failure. Intact fuel assemblies are stored in the spent fuel storage racks inside the fuel building.

The PWR fuel inventory is measured by the item accounting. Because the physical form of the fuel assembly dose not change before and after the burning in the core, there is no loss of fuel material in each transferring step. The new and spent fuel storage areas are continuously monitored by the CCTV. The IAEA inspection is regularly performed to trace the spent fuel movement in the spent fuel racks.

The rationles of PR assessment of a PWR Plant are as follows:

There are 2 types of PWR fuels, new and spent fuel, in a PWR Plant.

In this assessment, the fuel loaded in the reactor core is assumed as spent fuel from a conservative viewpoint.

- User Requirement 1.1

□ INPR1.1.1: Material quality

	New PWR fuel	Spent PWR fuel
Material type	LEU(3.5%)	Irradiated direct use material
Isotopic composition	$^{239}\text{Pu}/\text{Pu} = 0 \text{ wt}\%$,	$^{239}\text{Pu}/\text{Pu} = \sim 60 \text{ wt}\%$,
Radiation field (Dose rate of a fuel assembly)	Very low ($< 0.2 \text{ mSv/hr}$)	$\sim 10.37 \text{ Sv/hr}$
Heat generation	$^{238}\text{Pu}/\text{Pu} = 0 \text{ wt}\%$	$^{238}\text{Pu}/\text{Pu} = \sim 1.7 \text{ wt}\%$
Spontaneous neutron generation rate	$(^{240}\text{Pu} + ^{242}\text{Pu})/\text{Pu} = 0 \text{ wt}\%$	$(^{240}\text{Pu} + ^{242}\text{Pu})/\text{Pu} = \sim 30 \text{ wt}\%$

□ INPR1.1.2: Material quantity

	New PWR fuel	Spent PWR fuel
Mass of an item	$\sim 450 \text{ kg}$	$\sim 450 \text{ kg}$
No. of items to get one Significant quantity (SQ)	~ 4.5 assemblies because $\sim 2 \text{ ton-LEU}$ needed to get one SQ of ^{235}U	~ 2 assemblies because $\sim 0.9\text{MTHM}$ needed to get one SQ of Pu
No. of Significant Quantities which can be made from the fuel stored in the PWR plant	11 SQ On assumption that the new fuel storage are filled with 50 assemblies	375 SQ On assumption that the spent fuel storage and the reactor are filled with 750 assemblies.

□ INPR1.1.3: Material form

- ✓ No change of chemical form
- ✓ U: New PWR fuel assembly
- ✓ U, Pu: Spent PWR fuel assembly

□ Nuclear technology

- ✓ No enrichment process, no extraction of fissile materials and no target irradiation capability

Table 19. Evaluation of URPR1.1 of Step P5

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Material quality	Isotopic composition	²³⁹ Pu/ ²³⁹ Pu (wt%)	> 93	80~93	70~80	60~70	< 60
		²³⁵ U/ ²³⁵ U (wt%)	> 90	50~90	20~50	5~20	< 5
		²³² U/ ²³³ U (ppm) for ²³² U contamination	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	²³⁸ Pu/ ²³⁸ Pu (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	(²⁴⁰ Pu + ²⁴² Pu)/ ²³⁸ Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

- User Requirement 1.2

□ Accountability

- ✓ There is no MUF because of item accounting during irradiation in reactor and storing in spent fuel racks
- ✓ The NDA measurements systems is easily applied on the site by inspectors

□ Applicability of C/S measures

- ✓ Applicability of containment measures
The measurement system can be easily installed at the containment building and the fuel building.
- ✓ Applicability of surveillance measures

The surveillance system can be easily installed at the containment building and the fuel building..

✓ Applicability of other monitoring systems

The monitoring system can be easily installed and is operated under control of IAEA.

□ Detectability of nuclear material

✓ The new and spent PWR fuel are easily identified by NDA

✓ The radiation signature from the PWR spent fuel is hard due to its strong radioactivity

✓ The new and spent PWR fuel are easily identified by the passive system

□ Difficulty to modify the process

✓ Extent of automation

All the fuel loading and handling systems are not automated.

✓ Availability of data

The data is on-line transmitted to the operator.

✓ Authenticability of data

The data is very authentic because all the activities in the plant are open to IAEA.

✓ Transparency of process

All the activities in the plant are open to IAEA.

□ Difficulty to modify facility design

✓ It is very difficult to modify the fuel relevant facilities. Hot cell facility is required for treating the PWR spent fuel. The facility design is easily verified by inspectors.

Table 20. Evaluation of URPR1.2 of Step P5

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No			Yes	
	Applicability of surveillance measures		No			Yes	
	Applicability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Hardness of radiation signature		No reliable signature			Reliable signature	
	Need for passive/active mode		Active			Passive	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for Safeguards purpose		No			Yes	
	Transparency of process		No			Yes	
	Accessibility of material to inspectors for verification		No			Yes	
Difficulty to modify facility design	Verifiability of facility design by inspectors		No			Yes	

6.3.4 PR evaluation of “Transportation from interim storage of PWR spent fuel to DUPIC fuel fabrication plant” (Step D1)

The system characteristics of transportation of PWR spent fuel from interim storage to DUPIC fabrication plant are shown in Fig. 13 and described as followings.

The transportation activities consists of :

- Loading PWR spent fuel into transportation cask at the interim storage facility
- Transportation of the cask to the DUPIC facility by transport ship

- Unloading PWR spent fuel at the DUPIC facility

Considering the Korean situation, all inter-site transportation will be performed by ship and all intra-site movement will be performed by truck transport. The transportation system consists of transport casks, ship, trucks, etc.

During transport, the spent nuclear fuel is stored in shielding steel casks, each weighs about 80-120 ton with a capacity of 4-10 ton of fuel. The casks protect the surrounding area from dangerous radiation as well as preventing the fuel itself from being damaged. The cask are extremely sturdy, capable of withstanding a free fall from high height, immersion in deep water, violent fire and other situations without losing their integrity.

The ships have a number of unique design features to ensure that the highest levels of safety and reliability are maintained including:

- Duplication and high reliability electric power system
- Double hull construction with sub-division.

Electrical distribution systems based on two independent generators with 100% redundancy. Separated cargo holds with enhanced levels of radiation shielding, energy absorbing barriers, fire protection and radiation monitoring including

- Cargo cooling in each hold
- Extensive fire fighting equipment
- Satellite equipment for navigation, communication and ship location.

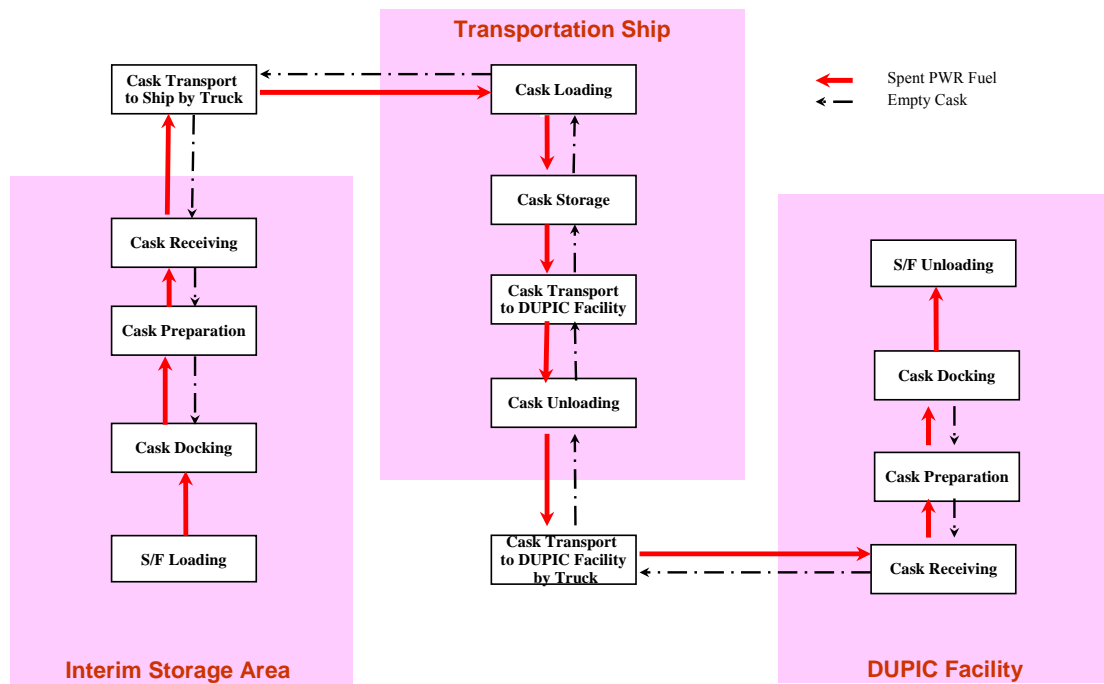


Figure 13. Work flow of PWR spent fuel transportation to DUPIC fabrication plant

The PR Characteristics of transportation of PWR spent fuel from interim storage to DUPIC fabrication plant are evaluated as followings.

Because the PWR spent fuel is very radioactive, the PWR spent fuel should be loaded in radiation shield cask and transported to the DUPIC facility without losing the original form as discharged from the PWR.

During the transportation, the spent nuclear fuel is item-counting and therefore MUF is zero. The plutonium content of the PWR spent fuels is ~0.7 wt% and the chemical form of the fuel is oxide.

The containment/surveillance equipments in all the entrance and exit areas of the interim storage facility and the DUPIC facility are installed and material flow are monitored.

During the transportation, only the permitted cask and path should be used, and the history of the PWR spent fuel and their containers are continuously traced, and reported to relevant authorities.

The radiation activity of the PWR spent fuel are still high, which makes it difficult to approach the nuclear material directly during transportation activities and also difficult to refurbish the loading and unloading facility for diversion and to install the diversion equipment in the facilities.

For the physical condition of the PWR spent fuel, the size and weight of the spent fuel assembly is $\sim 0.25 \text{ m} \times 0.25 \text{ m} \times 5 \text{ m}$ and $\sim 700 \text{ kg}$, respectively. The size of the cask is $\sim 2 \text{ m}$ in diameter and $\sim 6 \text{ m}$ in length. It should be noted that the size and weight of both the spent fuel assembly and cask are big and heavy.

The rationale of PR assessment of transportation of PWR spent fuel from interim storage to DUPIC fabrication plant are as followings:

- User Requirement 1.1
 - INPR1.1.1: Material quality
 - ✓ Material type: Irradiated direct use material
 - ✓ Isotopic composition: $^{239}\text{Pu}/\text{Pu} = \sim 60 \text{ wt}\%$
 - ✓ Radiation field: Dose rate of a fuel assembly is $\sim 1,037 \text{ rem/hr}$
 - ✓ Heat generation: $^{238}\text{Pu}/\text{Pu} = \sim 1.7 \text{ wt}\%$
 - ✓ Spontaneous neutron generation rate: $(^{240}\text{Pu} + ^{242}\text{Pu})/\text{Pu} = \sim 30 \text{ wt}\%$
 - INPR1.1.2: Material quantity
 - ✓ Mass of an item: $\sim 450 \text{ kg}$
 - ✓ No. of items to get one Significant quantity (SQ): ~ 2 assemblies because $\sim 0.9 \text{ MTHM}$ needed to get one SQ of Pu
 - ✓ No. of Significant Quantities which can be attained during the transportation of the PWR spent fuel assembly depends on the size of facility.
 - INPR1.1.3: Material form

- ✓ U, Pu: PWR spent fuel with medium burnup (35,000 MWd/t)
- Nuclear technology
 - ✓ The whole process employs handling by assembly or cask without losing the original form as discharged from the nuclear power plant.

Table 21. Evaluation of URPR1.1 of Step D1

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Material quality	Isotopic composition	$^{239}\text{Pu}/\text{Pu}$ (wt%)	> 93	80~93	70~80	60~70	< 60
		$^{235}\text{U}/\text{U}$ (wt%)	> 90	50~90	20~50	5~20	< 5
		$^{232}\text{U}_{\text{contam. for } ^{233}\text{U}}$ (ppm)	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	$^{238}\text{Pu}/\text{Pu}$ (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	($^{240}\text{Pu} + ^{242}\text{Pu}$)/Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

- User Requirement 1.2
 - Accountability
 - ✓ MUF: 0
 - ✓ Measurement method/equipment

The item accounting for transported PWR spent fuel assembly shall be applied at the shipping area and receiving area. The weighing and

NDA systems for the transported assembly accounting would be applied at the shipping area and receiving area.

- Applicability of C/S measures
 - ✓ The containment and surveillance systems can be easily installed at the shipping area and receiving area.
- Applicability of other monitoring systems
 - ✓ Feed material measurement: PWR spent fuel assembly scanning system
- Detectability of nuclear material
 - ✓ The PWR spent fuel are easily identified by NDA during the transportation
 - ✓ The radiation signature from the PWR spent fuel is hard due to its strong radioactivity
 - ✓ The PWR spent fuel are easily identified by the passive system
- Difficulty to modify the process
 - ✓ Extent of automation

All the fuel loading and handling systems during transportation are not automated.
 - ✓ Availability of data

The data is checked by the operator before and after transportation.
 - ✓ Authenticability of data

The data is very authentic because all the activities during transportation are open to IAEA.
 - ✓ Transparency of process

All the activities during transportation are open to IAEA.
- Difficulty to modify facility design
 - ✓ It is very difficult to modify the transportation equipment. Hot cell facility is required for treating the PWR spent fuel. The transportation equipment is easily verified by inspectors.

Table 22. Evaluation of URPR1.2 of Step D1

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No			Yes	
	Applicability of surveillance measures		No			Yes	
	Applicability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Hardness of radiation signature		No reliable signature			Reliable signature	
	Need for passive/active mode		Active			Passive	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for safeguards purpose		No			Yes	
	Transparency of process		No			Yes	
	Accessibility of material to inspectors for verification		No			Yes	
Difficulty to modify facility design	Verifiability of facility design by inspectors		No			Yes	

6.3.5 PR evaluation of “DUPIC fuel fabrication plant” (Step D2)

The system characteristics of DUPIC fuel fabrication plant are as followings.

The DUPIC fuel fabrication process involves the direct refabrication of PWR spent fuel for a CANDU fuel. The spent fuel materials are recovered from the PWR spent fuel by disassembling and decladding using only thermal and mechanical processes.

The powder preparation process called OREOX (Oxidation REDuction of OXide fuel) is considered the most critical process for producing resinterable powder feedstock. Once

the resinterable powder is prepared, the pellet and rod manufacturing processes are almost same as the conventional powder/pellet route in CANDU fuel fabrication.

The facility parameters are,

- Capacity: 178 MtHM/year (assuming DUPIC fuel cycle of 10 GWe/year)
- Reference PWR spent fuel: 35,000 MWd/MtU with 10 years cooling

The PR characteristics of DUPIC fabrication plant are evaluated as followings.

Due to the dry process, no fissile material can be separated in pure form. The material requires further chemical reprocessing in order to obtain material suitable for weapon purpose.

The presence of some fission products leads to a high dose rate arising from the material. The DUPIC process has to be carried out in heavily shielded hot cell because it handles its highly radioactive materials. The processing is self-contained, and there is no transport of intermediate materials outside the facility. Therefore, access to the nuclear materials is extremely difficult.

The rationale of PR assessment of DUPIC fabrication plant are as followings.

- User Requirement 1.1
 - INPR1.1.1: Material quality
 - ✓ Material type: Irradiated direct use material
 - ✓ Isotopic composition: $^{239}\text{Pu}/\text{Pu} \approx 60 \text{ wt}\%$
 - ✓ Radiation field: Dose rate of a fuel bundle is $\sim 15 \text{ rem/hr}$
 - ✓ Heat generation: $^{238}\text{Pu}/\text{Pu} = 1.7 \text{ wt}\%$
 - ✓ Spontaneous neutron generation rate: $(^{240}\text{Pu} + ^{242}\text{Pu})/\text{Pu} \approx 30 \text{ wt}\%$
 - INPR1.1.2: Material quantity
 - ✓ Mass of an item: $\sim 24 \text{ kg}$
 - ✓ No. of items to get one Significant quantity (SQ): ~ 48 assemblies because $\sim 0.9 \text{ MTHM}$ is needed to make one SQ of Pu from the DUPIC fuel

- ✓ No. of Significant Quantities which can be attained during the DUPIC fuel fabrication process
- INPR1.1.3: Material form
 - ✓ U, Pu: PWR spent fuel with medium burnup (35,000 MWd/t)
- Nuclear technology
 - ✓ The whole process employs only the thermal and mechanical processes and there is no chemical process. Therefore, it is impossible to extract fissile materials and to modify the DUPIC fuel cycle facility and processes for enrichment.

Table 23. Evaluation of URPR1.1 of Step D2

Indicators	Evaluation Parameter		Evaluation scale				
			W		S		
			VW	W	M	S	VS
Material quality	Isotopic composition	²³⁹ Pu/Pu (wt%)	> 93	80~93	70~80	60~70	< 60
		²³⁵ U/U (wt%)	> 90	50~90	20~50	5~20	< 5
		²³² U _{contam.} for ²³³ U (ppm)	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	²³⁸ Pu/Pu (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	(²⁴⁰ Pu+ ²⁴² Pu)/Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

- User Requirement 1.2
 - Accountability

- ✓ MUF: $4.01 \text{ kg Pu} = 178 \text{ (tHM)} \times 0.009 \text{ (Pu/HM)} \times 0.01 \text{ (error)} \times 0.25 \text{ (3 months)}$
- ✓ Measurement method/equipment
A Near Real Time Accounting System (NRTA) for fissile accountability system shall be used in the plant. The NRTA system is integrated with individual nuclear material measurement system. The item accounting for both PWR incoming fuel and outgoing DUPIC fuel shall be based on the modified curium counter. The weighing and NDA systems for the bulk accounting in DUPIC process would be applied.
- Applicability of C/S measures
 - ✓ The contaminant and surveillance systems can be easily installed at the hot cell facility
- Applicability of other monitoring systems
 - ✓ Feed material measurement: PWR spent fuel rod scanning system
 - ✓ Process monitoring: Unattended continuous hot-cell monitoring system
- Detectability of nuclear material
 - ✓ The nuclear material in the fuel fabrication facility is easily identified by NDA
 - ✓ The radiation signature during fuel fabrication process is hard due to its strong radioactivity
 - ✓ The nuclear material during fuel fabrication process is easily identified by the passive system
- Difficulty to modify the process
 - ✓ Extent of automation
Some of fabrication processes would not be automated.
 - ✓ Availability of data
The data is on-line transmitted to the operator.
 - ✓ Authenticability of data
The data is very authentic because all the activities in the fabrication facility are open to IAEA.
 - ✓ Transparency of process
All the activities in the fabrication facility are open to IAEA.

- Difficulty to modify facility design
 - ✓ It is very difficult to modify the fuel relevant facilities. Hot cell facility is required for treating the PWR spent fuel. The facility design is easily verified by inspectors.

Table 24. Evaluation of URPR1.2 of Step D2

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No			Yes	
	Applicability of surveillance measures		No			Yes	
	Applicability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Hardness of radiation signature		No reliable signature			Reliable signature	
	Need for passive/active mode		Active			Passive	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for safeguards purpose		No			Yes	
	Transparency of process		No			Yes	
	Accessibility of material to inspectors for verification		No			Yes	
Difficulty to modify facility design	Verifiability of facility design by inspectors		No			Yes	

Considering the Basic Principle 2 regarding the robustness and multiple barriers, etc, the pathway analysis needs to be utilized. The diversion path and barriers of DUPIC fabrication can be considered in viewpoints of Acquisition, Processing and Fabrication of nuclear weapon as shown in Fig. 14.

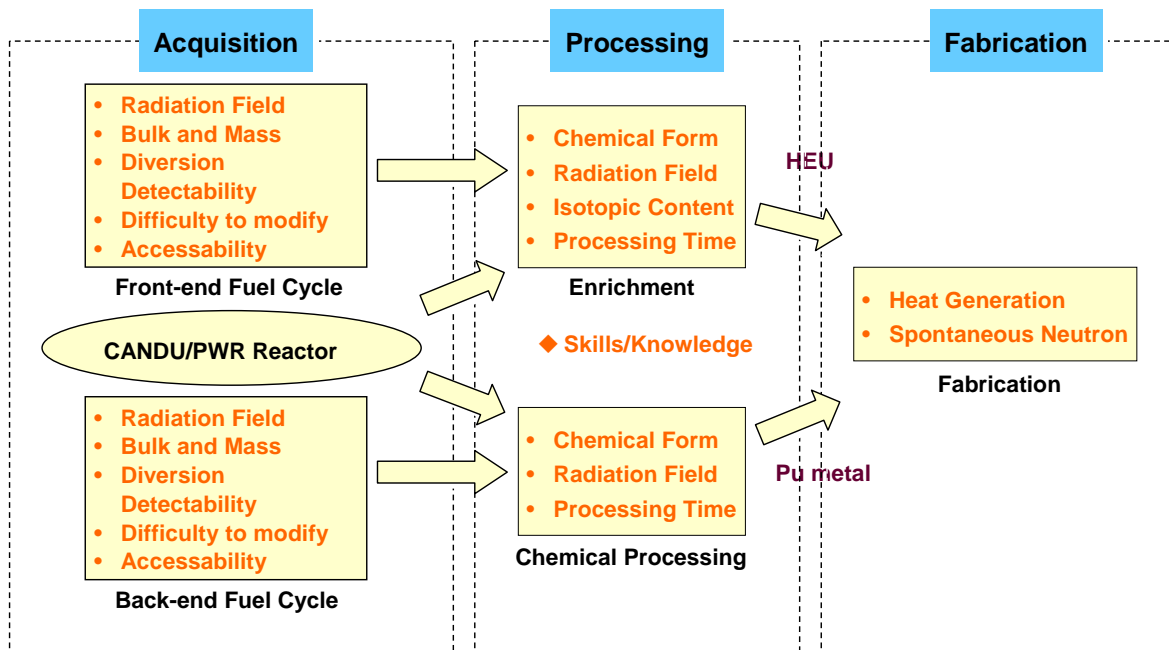


Figure 14. Diversion path and barriers in case of DUPIC fabrication

In INPRO methodology, physical protection is not considered for the diversion. And it will be assumed that the facility owner is the State, and so the proliferator will be an insider. Hence, the acquisition of nuclear materials from a facility will be successful by concealing the State intention under the IAEA. This implies only covert diversion by national government as shown in Table 25.

Table 25. Diversion type correspondence to proliferators

Proliferators	Covert diversion	Overt diversion
Sub-national Group	X	X
National government	O	X

In order to assess Basic Principle 2, the analysis of the acquisition paths of diversion of nuclear material in DUPIC fabrication plant can be considered as shown in Fig. 15 and Fig. 16.

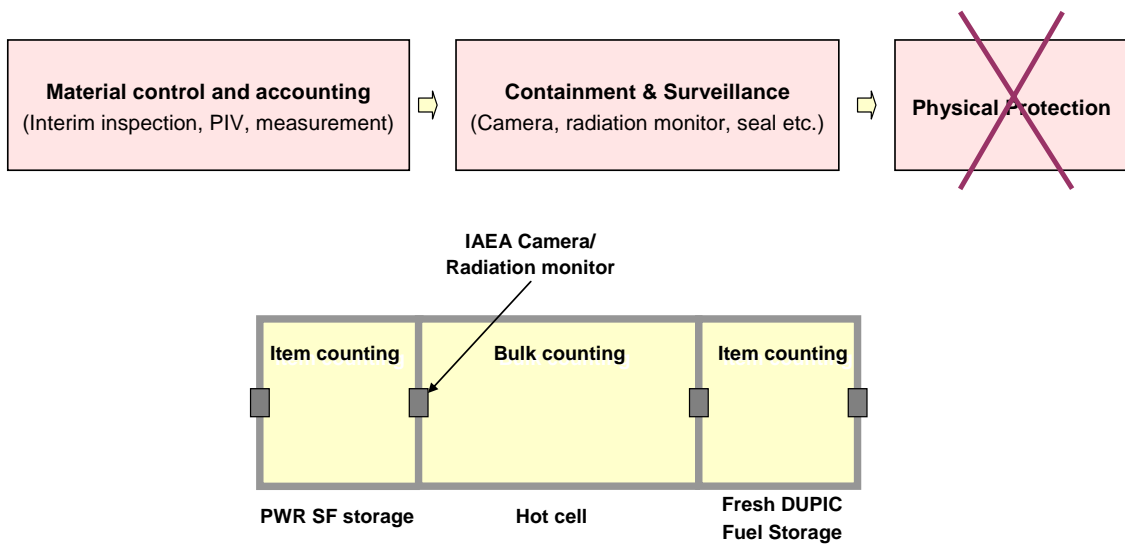


Figure 15. Acquisition path of nuclear material in DUPIC fabrication plant

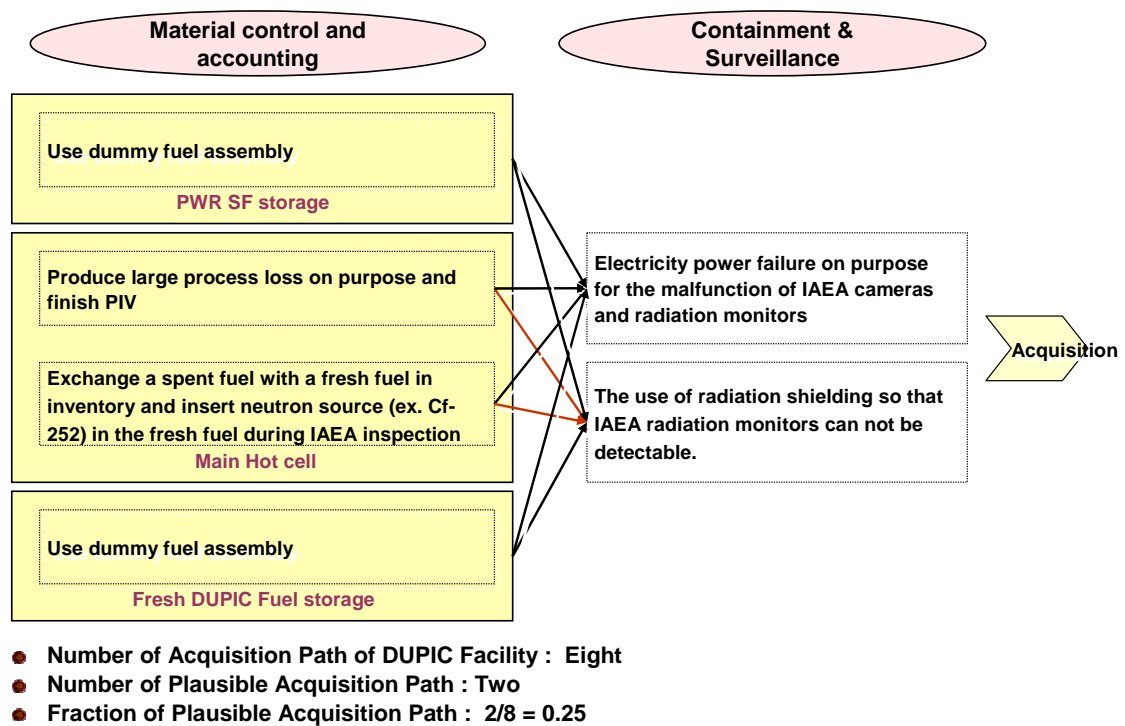


Figure 16. Plausible acquisition paths of nuclear material in DUPIC fabrication plant

- User Requirement 2.1

- INPR2.1.1: The extent by which the INS is covered by multiple intrinsic features

For DUPIC fabrication plant, while the number of hypothetical acquisition paths is 8, the plausible paths is 2. Two plausible paths are covered by multiple barriers. Therefore, the ratio of “Number of plausible acquisition paths covered by multiple PR features and measures” to “Number of plausible acquisition paths” is 1.

- INPR2.1.2: Robustness of barriers covering an acquisition path

To evaluate the robustness of barriers covering the plausible acquisition path for the DUPIC fuel fabrication, it is needed to aggregate the barriers for the acquisition path. But, the aggregation method was not setup yet in this study.

- User Requirement 2.2

- INPR2.2.1: Cost to incorporate those intrinsic features and extrinsic measures, which are required to provide PR

Cost effectiveness of the PR for the system was not evaluated in this study.

Table 26. Evaluation of URPR2.1 of Step D2

Indicators	Evaluation Parameter	Evaluation scale				
		VW	W	M	S	VS
INPR2.1.1: The extent by which the INS is covered by multiple intrinsic features	“No. of plausible acquisition paths covered by multiple PR features and measures” and “No. of plausible acquisition paths”	< 0.2	0.2 ~ 0.4	0.4 ~ 0.6	0.6 ~ 0.8	> 0.8
INPR2.1.2: Robustness of barriers covering an acquisition path	Extent of robustness of barriers	Very little	Little	Medium	Great	Very great

6.3.6 PR evaluation of “CANDU plant with DUPIC fuel” (Step D4)

The system characteristics of CANDU plant are as followings.

If the DUPIC fuel is arrived at the CANDU plant, fuel bundles are counted and stored in the fuel racks residing at the bottom of the storage bay. The DUPIC fuel bundles (eight bundles per day) are subject to the visual inspection and dimension measurement before loading. A fuel manipulator transports the DUPIC fuel bundle from the fuel rack to the conveyor. The DUPIC fuel bundle will be loaded to the CANDU reactor core in the reverse way of discharging path of spent CANDU fuel. When the fuel bundle is transferred to the currently known as discharge bay, the fuel elevator places the fuel bundles in the fueling machine. The fuel bundles are loaded into the fuel channels selected by the operator. The average fuel residence time in the core is 610 days. The fissile content of the DUPIC fuel is 1.5 wt% when the fuel loaded, while it is 0.7 wt% when discharged.

The integrity of the fuel during the normal operation is monitored by the radiation level of the coolant. As new fuel bundles are loaded, the burnt DUPIC fuel bundles are discharged from the core and automatically transferred to the reception bay. The spent fuels are inspected for the failure. Intact fuel bundles are moved from the discharge bay to the storage bay.

The PR characteristics of CANDU plant are evaluated as followings.

The DUPIC fuel inventory is measured by the item counting. Because the physical form of the fuel bundle dose not change before and after the burning in the core, there is no loss of fuel material in each transferring step. The spent fuel bay is continuously monitored by the CCTV. IAEA inspection is regularly performed to trace the spent fuel movement in the spent fuel bay.

The DUPIC fuel bundles are remotely and automatically loaded into the core and discharged from the core. There is no need for the operator or any other person to physically handle the fuel. The integrity of the fuel is monitored by the radiation activity when the fuel channel is open for refueling or inspection.

Dummy fuel bundles may be used for the maintenance of the fueling machine and system. During the normal and abnormal operations, there is no way that the fuel bundles are repositioned without using the fueling machine. Passages to the reactor building are the equipment lock and spent fuel transfer canal.

The rationale of PR assessment of CANDU plant loaded with DUPIC fuel are as followings.

- User Requirement 1.1
 - INPR1.1.1: Material quality
 - ✓ Material type: Irradiated direct use material
 - ✓ Isotopic composition: $^{239}\text{Pu}/\text{Pu} = 43 \sim 63 \text{ wt\%}$, $^{235}\text{U}/\text{U} = 0.3\text{-}1.0 \text{ wt\%}$, $^{232}\text{U}/^{233}\text{U} = \sim 1300 \text{ ppm}$
 - ✓ Radiation field: Dose rate of a fuel bundle is $15 \sim 61 \text{ rem/hr}$
 - ✓ Heat generation: $^{238}\text{Pu}/\text{Pu} = 1.5 \sim 1.9 \text{ wt\%}$
 - ✓ Spontaneous neutron generation rate: $(^{240}\text{Pu} + ^{242}\text{Pu})/\text{Pu} = 30 \sim 46 \text{ wt\%}$
 - INPR1.1.2: Material quantity
 - ✓ Mass of an item: $\sim 24 \text{ kg}$
 - ✓ No. of items to get one Significant quantity (SQ): ~ 52 assemblies because $\sim 1 \text{ MTHM}$ is needed to make one SQ of Pu from the DUPIC spent fuel
 - ✓ No. of Significant Quantities which can be attained during the CANDU plant operation.
 - INPR1.1.3: Material form
 - ✓ No change of chemical form
 - Nuclear technology
 - ✓ No enrichment process, no extraction of fissile materials and no target irradiation capability

Table 27. Evaluation of URPR1.1 of Step D4

Indicators	Evaluation Parameter		Evaluation scale				
			W		S		
			VW	W	M	S	VS
Material quality	Isotopic composition	$^{239}\text{Pu}/\text{Pu}$ (wt%)	> 93	80~93	70~80	60~70	< 60
		$^{235}\text{U}/\text{U}$ (wt%)	> 90	50~90	20~50	5~20	< 5
		$^{232}\text{U}_{\text{contam.}}$ for ^{233}U (ppm)	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	$^{238}\text{Pu}/\text{Pu}$ (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	($^{240}\text{Pu} + ^{242}\text{Pu}$)/Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

- User Requirement 1.2

□ Accountability

- ✓ There is no MUF because of item accounting during irradiation in reactor

□ Applicability of C/S measures

- ✓ Applicability of containment measures

The measurement system can be easily installed at the containment building.

- ✓ Applicability of surveillance measures

The surveillance system can be easily installed.

- ✓ Applicability of other monitoring systems

The monitoring system can be easily installed.

- Detectability of nuclear material
 - ✓ The new and spent DUPIC fuel are easily identified by NDA
 - ✓ The radiation signature from the new and spent DUPIC fuel is hard due to its strong radioactivity
 - ✓ The new and spent DUPIC fuel are easily identified by the passive system
- Difficulty to modify the process
 - ✓ Extent of automation

The fuel loading system is fully automated. Others are partially automated.
 - ✓ Availability of data

The data is on-line transmitted to the operator.
 - ✓ Authenticability of data

The data is very authentic because all the activities in the plant are open to IAEA.
 - ✓ Transparency of process

The DUPIC fuel stays in the shipping cask, water pool, reactor building and reactor vessel.
- Difficulty to modify facility design
 - ✓ Verifiability of facility design by inspectors: Hot cell is required for the radiation protection.

Table 28. Evaluation of URPR1.2 of Step D4

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No			Yes	
	Applicability of surveillance measures		No			Yes	
	Applicability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Hardness of radiation signature		No reliable signature			Reliable signature	
	Need for passive/active mode		Active			Passive	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for safeguards purpose		No			Yes	
	Transparency of process		No			Yes	
	Accessibility of material to inspectors for verification		No			Yes	
Difficulty to modify facility design	Verifiability of facility design by inspectors		No			Yes	

In order to assess Basic Principle 2, the analysis of acquisition path of diversion of nuclear material in CANDU plant needs to be considered. At first, safeguards approach of a CANDU plant should be assessed.

Verification of fresh fuel

Verification of fresh fuel is the number of item and serial number. Verification of nuclear fuel is used by HM-4 or HM-5. The Fieldspec (HM-4, -5) as shown in Fig. 17, portable sodium iodine based instrument, can determine the presence of radioactive materials. It is used for both inspection verification activities and

complementary access. It also has direct application in searches for indications of the illicit trafficking of nuclear material.

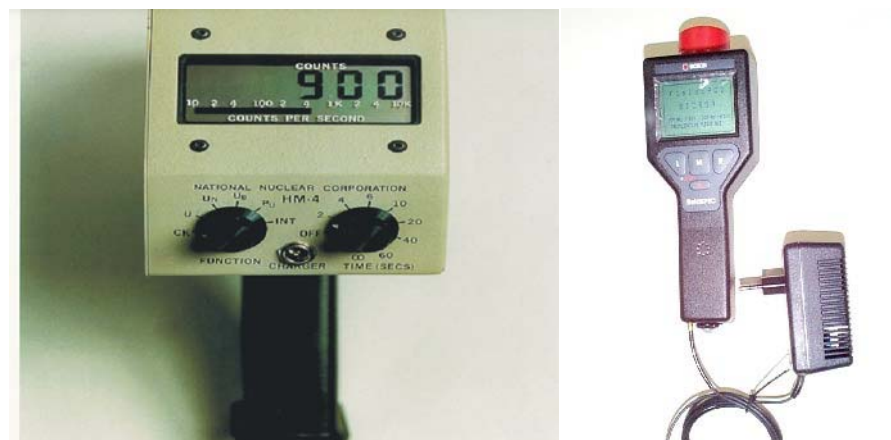


Figure 17. Verification tools of nuclear fuel

Verification of fuel in reactor

For the verification of CANDU reactor fuels, VIFM (VXI Integrated Fuel Monitor) has been used as shown in Fig. 18. These play a key role in the spent fuel monitoring of on-load refueled reactors. The VIFM is including two CDM (Core Discharge Monitor) and Two BCs (Bundle Counter). The CDM installed on the bottom of reactor building consists of ion chamber for gamma and fission chamber for neutron, and can recognize the fuel transfer through the fuel loading machine. The BC can recognize transfer of spent fuel to the reception bay and their number. The inspector can verify the amount and number of discharged SF using VIFM data provided by operator.

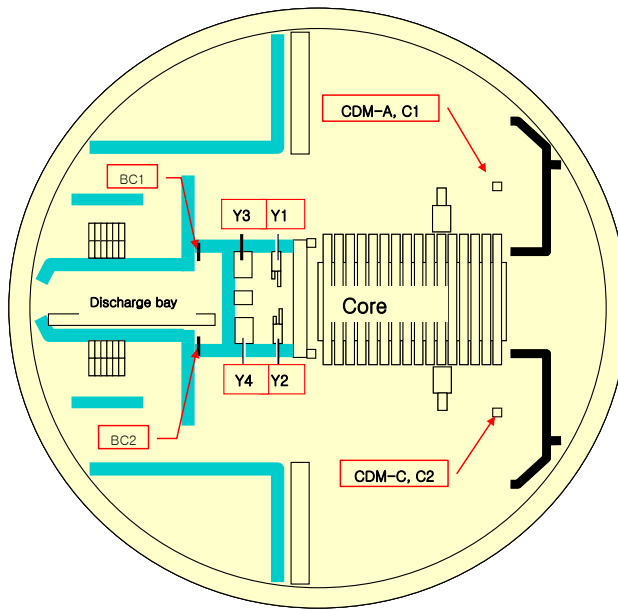


Figure 18. Verification of fuel in reactor

DUPIC Fuel Loading Route

The use of a spent fuel discharge path in the reverse direction has been proposed as an alternative to the current new-fuel loading path. Once fresh DUPIC fuel is transported to the plant, it will be transferred to the reactor core as following procedures:

- ① The fresh fuel, which is in the transportation cask, is moved to the fuel basket in the welding station.
- ② The fuel basket is lowered into the storage bay.
- ③ The tilt mechanism changes the vertical orientation of the fuel basket to a horizontal one.
- ④ The fuel bundles are located in the tray (or fuel rack).
- ⑤ The fuel bundles are transferred to the discharge bay.
- ⑥ The spent-fuel elevator operates in the reverse direction to move the fresh DUPIC fuel to the currently known as discharge port.
- ⑦ The fuel bundles are moved into the fueling machine.
- ⑧ The fresh DUPIC fuels are automatically located in the fuel channel and spent DUPIC fuel is discharged.
- ⑨ The DUPIC spent fuel follows the existing discharge route.

Diversion Scenario During DUPIC Fuel Loading

During DUPIC loading, it is difficult to distinguish the fresh DUPIC fuel and DUPIC spent fuel by the CDMs as shown in Fig. 20 because the both fuels are emitting neutron as well as gamma radiation. At discharge bay, the bundle counter can not distinguish the direction in which a fresh fuel or a spent fuel is going up or down.

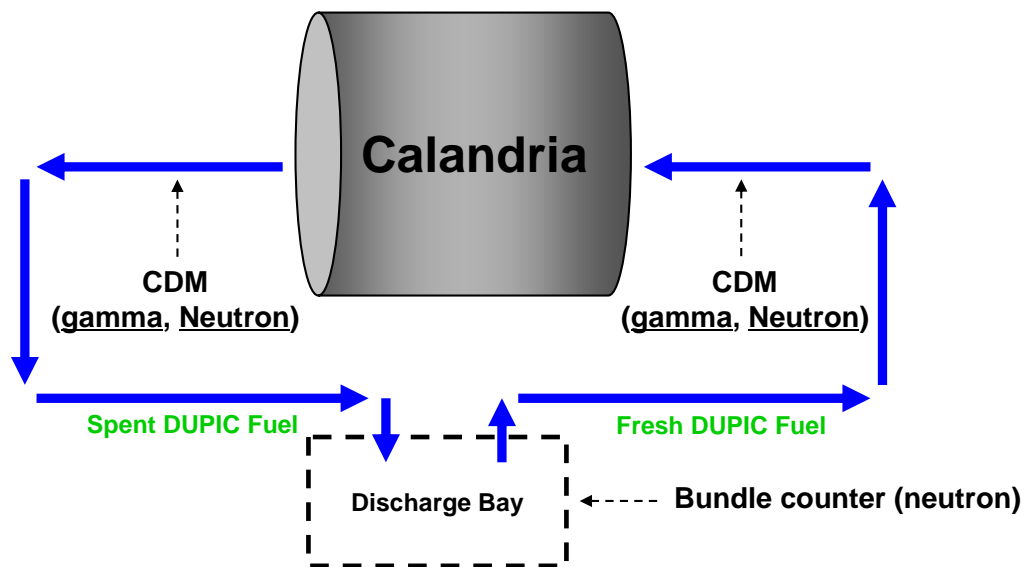
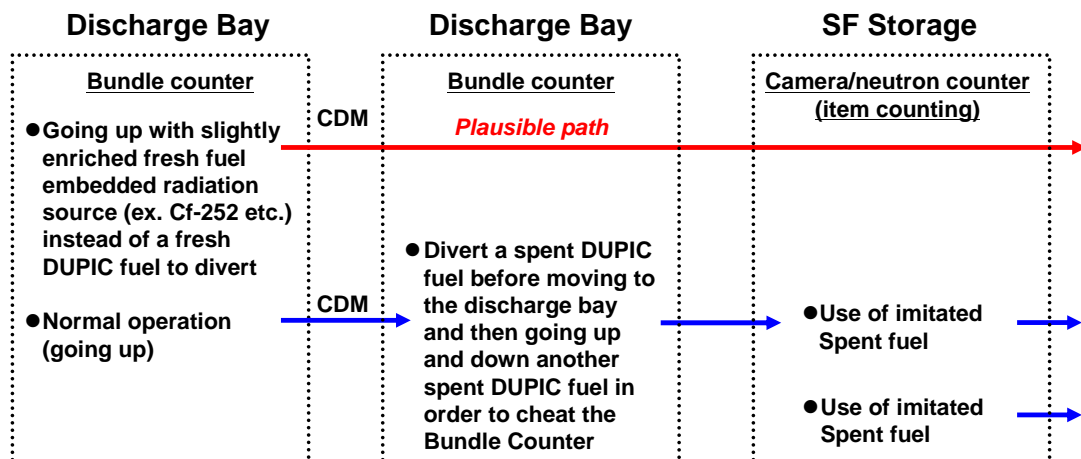


Figure 19. Diversion scenario during DUPIC fuel loading

Diversion Scenario at CANDU Plant

Diversion scenario were shown in Figure 20. The acquisition paths during discharging DUPIC spent fuel may be hypothesized as three paths to divert the DUPIC spent fuel.



**Number of the plausible acquisition path of DUPIC fuel
in CANDU reactor : one (red color)**

Figure 20. Hypothetical diversion scenarios in CANDU reactor

- User Requirement 2.1

- INPR2.1.1: The extent by which the INS is covered by multiple intrinsic features

- ✓ For CANDU plant, the number of hypothetical acquisition paths is 3. And the number of plausible acquisition path among the hypothetical acquisition paths is 1 and it is covered by multiple barriers. Therefore, the ratio of “Number of plausible acquisition paths covered by multiple PR features and measures” to “Number of plausible acquisition paths” is 1.

- INPR2.1.2: Robustness of barriers covering an acquisition path

- ✓ The robustness of barriers covering the plausible acquisition path for a DUPIC fuel in CANDU plant needs aggregation method to integrate the barriers for the acquisition path. But, the aggregation method was not setup yet in this study.

- User Requirement 2.2

- INPR2.2.1: Cost to incorporate those intrinsic features and extrinsic measures, which are required to provide PR

- ✓ Cost effectiveness of the PR for the system was not evaluated in this study.

Table 29. Evaluation of URPR2.1 of Step D4

Indicators	Evaluation Parameter	Evaluation scale				
		VW	W	M	S	VS
INPR2.1.1: The extent by which the INS is covered by multiple intrinsic features	“No. of plausible acquisition paths covered by multiple PR features and measures” and “No. of plausible acquisition paths”	< 0.2	0.2 ~ 0.4	0.4 ~ 0.6	0.6 ~ 0.8	> 0.8
INPR2.1.2: Robustness of barriers covering an acquisition path	Extent of robustness of barriers	Very little	Little	Medium	Great	Very great

6.3.7 PR evaluation of “CANDU NU fuel cycle”

The standard CANDU fuel cycle can be easily constructed from those of the whole DUPIC fuel cycle concept. The front end of the CANDU fuel cycle is the same as that of the PWR fuel cycle except for the enrichment process. The backend of the CANDU fuel cycle is the same as that of the DUPIC fuel cycle. The difference between the standard CANDU and DUPIC fuel cycle comes from the fuel composition. As it was postulated for the DUPIC fuel cycle analysis, it is not necessary to consider the fuel cycle components prior to the CANDU fuel fabrication because those activities are not performed in the Korean peninsula. The evaluation of each fuel cycle component was performed by considering the difference of the fuel properties between the standard CANDU NU and DUPIC fuels, and their evaluation results are shown in the following sections.

6.3.7.1 PR evaluation of “CANDU NU fuel fabrication plant”

Table 30. Evaluation of URPR1.1 of CANDU NU fuel fabrication plant (C3)

Indicators	Evaluation Parameter		Evaluation scale				
			W		S		
			VW	W	M	S	VS
Material quality	Isotopic composition	²³⁹ Pu/Pu (wt%)	> 93	80~93	70~80	60~70	< 60
		²³⁵ U/U (wt%)	> 90	50~90	20~50	5~20	< 5
		²³² U _{contam.} for ²³³ U (ppm)	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	²³⁸ Pu/Pu (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	(²⁴⁰ Pu+ ²⁴² Pu)/Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

Table 31. Evaluation of URPR1.2 of CANDU NU fuel fabrication plant (C3)

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No			Yes	
	Applicability of surveillance measures		No			Yes	
	Applicability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Hardness of radiation signature		No reliable signature			Reliable signature	
	Need for passive/active mode		Active			Passive	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for safeguards purpose		No			Yes	
	Transparency of process		No			Yes	
	Accessibility of material to inspectors for verification		No			Yes	
Difficulty to modify facility design	Verifiability of facility design by inspectors		No			Yes	

6.3.7.2 Evaluation of “Transportation of NU fuel to the CANDU plant”

Table 32. Evaluation of URPR1.1 of transportation of NU fuel to the CANDU plant(C4)

Indicators	Evaluation Parameter		Evaluation scale				
			W		S		
			VW	W	M	S	VS
Material quality	Isotopic composition	$^{239}\text{Pu}/\text{Pu}$ (wt%)	> 93	80~93	70~80	60~70	< 60
		$^{235}\text{U}/\text{U}$ (wt%)	> 90	50~90	20~50	5~20	< 5
		$^{232}\text{U}_{\text{contam. for } ^{233}\text{U}}$ (ppm)	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	$^{238}\text{Pu}/\text{Pu}$ (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	($^{240}\text{Pu} + ^{242}\text{Pu}$)/Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

Table 33. Evaluation of URPR1.2 of transportation of NU fuel to the CANDU plant(C4)

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No			Yes	
	Applicability of surveillance measures		No			Yes	
	Applicability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Hardness of radiation signature		No reliable signature			Reliable signature	
	Need for passive/active mode		Active			Passive	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for Safeguards purpose		No			Yes	
	Transparency of process		No			Yes	
	Accessibility of material to inspectors for verification		No			Yes	
Difficulty to modify facility design	Verifiability of facility design by inspectors		No			Yes	

6.3.7.3 PR evaluation of “CANDU plant with CANDU NU fuel”

Table 34. Evaluation of URPR1.1 of CANDU plant with CANDU NU fuel (C5)

Indicators	Evaluation Parameter		Evaluation scale				
			W		S		
			VW	W	M	S	VS
Material quality	Isotopic composition	²³⁹ Pu/Pu (wt%)	> 93	80~93	70~80	60~70	< 60
		²³⁵ U/U (wt%)	> 90	50~90	20~50	5~20	< 5
		²³² U _{contam.} for ²³³ U (ppm)	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	²³⁸ Pu/Pu (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	(²⁴⁰ Pu+ ²⁴² Pu)/Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

Table 35. Evaluation of URPR1.2 of CANDU plant with CANDU NU fuel (C5)

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No			Yes	
	Applicability of surveillance measures		No			Yes	
	Applicability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Hardness of radiation signature		No reliable signature			Reliable signature	
	Need for passive/active mode		Active			Passive	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for safeguards purpose		No			Yes	
	Transparency of process		No			Yes	
	Accessibility of material to inspectors for verification		No			Yes	
Difficulty to modify facility design	Verifiability of facility design by inspectors		No			Yes	

6.3.7.4 PR evaluation of “Permanent disposal of spent CANDU NU fuel”

Table 36. Evaluation of URPR1.1 of permanent disposal of spent CANDU NU fuel(C7)

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Material quality	Isotopic composition	²³⁹ Pu/Pu (wt%)	> 93	80~93	70~80	60~70	< 60
		²³⁵ U/U (wt%)	> 90	50~90	20~50	5~20	< 5
		²³² U _{contam.} for ²³³ U (ppm)	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	²³⁸ Pu/Pu (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	(²⁴⁰ Pu+ ²⁴² Pu)/Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

Table 37. Evaluation of URPR1.2 of permanent disposal of spent CANDU NU fuel(C7)

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No			Yes	
	Applicability of surveillance measures		No			Yes	
	Applicability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Hardness of radiation signature		No reliable signature			Reliable signature	
	Need for passive/active mode		Active			Passive	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for safeguards purpose		No			Yes	
	Transparency of process		No			Yes	
	Accessibility of material to inspectors for verification		No			Yes	
Difficulty to modify facility design	Verifiability of facility design by inspectors		No			Yes	

6.3.8 PR evaluation of “Permanent disposal” (Step D8)

The system characteristics of permanent disposal of DUPIC spent fuel are as followings.

The disposal facility is assumed as follows.

- Size: 11,000 MtHM, 500 MtHM/year
- Operation/Maintenance: container emplacement for 22 years, closure work for 5 year, monitoring for 300 years

The disposal facility consists of two parts: the surface facility and the underground facility. The reference disposal facility is assumed to be the room-and-pillar configuration for the underground excavations which consists of a series of regularly spaced disposal rooms and connecting tunnels.

The spent fuel bundles are sealed into containers in fuel packaging facility before they are transported to the disposal vault or temporary storage area. The disposal vault is reached and serviced by shafts. The containers are transported into the underground facilities and are placed into vertical boreholes drilled into the floor of the disposal rooms. The container is surrounded by the clay-based buffer material within each borehole. Each disposal room is backfilled with clay-based backfill materials, and the room entrance is sealed when all of the boreholes have been filled (as shown in Fig. 21).

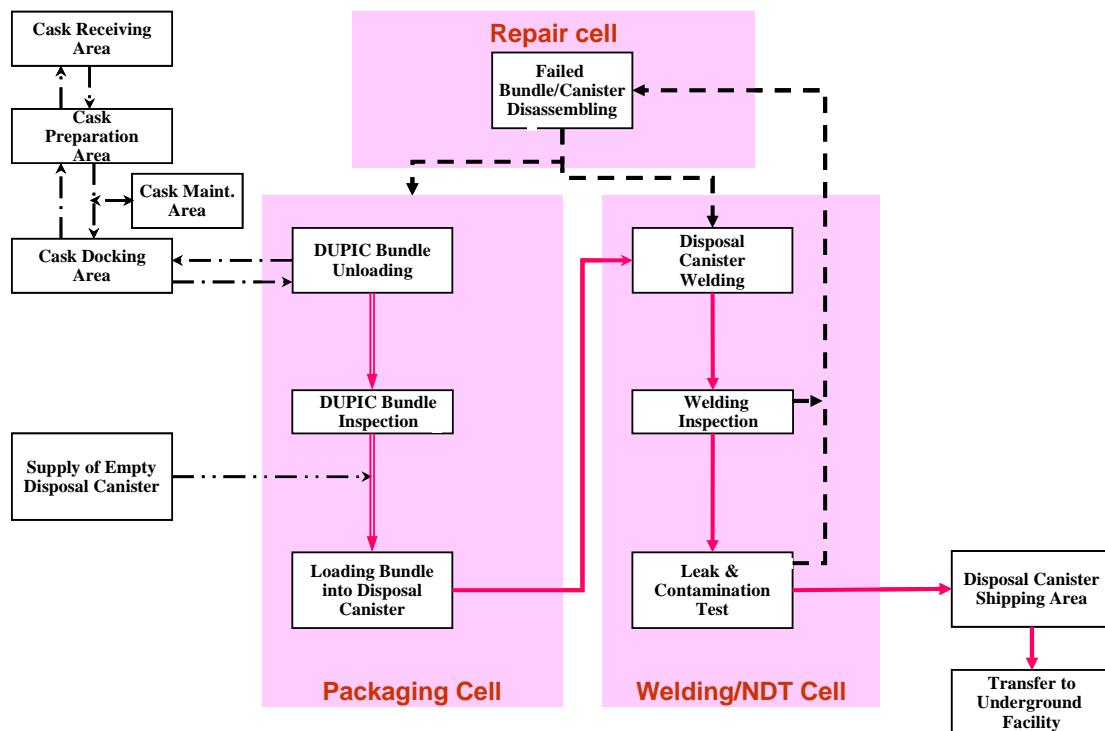


Figure 21. Work flow for permanent disposal of the DUPIC spent fuel

The PR characteristics of a permanent disposal of DUPIC spent fuel are evaluated as followings.

Permanent disposal is item-counting facility and therefore MUF is zero. The plutonium content of the DUPIC spent fuels is ~0.4 wt% and the chemical form of the fuel is oxide.

During operation, the history of the DUPIC spent fuel and their containers are continuously traced. The containment/surveillance equipment at all the entrance and exit areas of the disposal facility are installed and material flow are monitored. It is extremely difficult to contact the nuclear material directly during disposal activities and also difficult to refurbish the packaging and disposal facility for diversion and to install diversion equipment in the disposal facility. But the radiation field gradually becomes weaker with time, and after hundreds of years, the radiation field will be no longer active.

The rationale of PR assessment of a permanent disposal are as followings.

- User Requirement 1.1
 - INPR1.1.1: Material quality
 - ✓ Material type: Irradiated direct use material
 - ✓ Isotopic composition: $^{239}\text{Pu}/\text{Pu} = \sim 40 \text{ wt}\%$
 - ✓ Radiation field: Dose rate of a fuel bundle is $\sim 61 \text{ rem/hr}$
 - ✓ Heat generation: $^{238}\text{Pu}/\text{Pu} = \sim 4.9 \text{ wt}\%$
 - ✓ Spontaneous neutron generation rate: $(^{240}\text{Pu} + ^{242}\text{Pu})/\text{Pu} = \sim 49 \text{ wt}\%$
 - INPR1.1.2: Material quantity
 - ✓ Mass of an item: $\sim 24 \text{ kg}$
 - ✓ No. of items to get one Significant quantity (SQ): ~ 52 assemblies because $\sim 1 \text{ MTHM}$ is needed to make one SQ of Pu from the DUPIC fabrication
 - ✓ No. of Significant Quantities which can be attained during the permanent disposal of DUPIC spent fuel.
 - INPR1.1.3: Material form
 - ✓ U, Pu: DUPIC spent fuel with low burnup ($15,000 \text{ MWd/t}$)

□ Nuclear technology

- ✓ The whole process employs only handling by bundle or cask without losing the original form as discharged from the CANDU nuclear power plant.

Table 38. Evaluation of URPR1.1 of Step D8

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Material quality	Isotopic composition	²³⁹ Pu/Pu (wt%)	> 93	80~93	70~80	60~70	< 60
		²³⁵ U/U (wt%)	> 90	50~90	20~50	5~20	< 5
		²³² U _{contam.} for ²³³ U (ppm)	< 1	1~100	100~4000	4000~7000	> 7000
	Material type		DUM	DIM	L	N	D
	Radiation field	Dose (mSv/hr)	< 10	10~150	150~1000	1000~10000	> 10000
	Heat generation	²³⁸ Pu/Pu (wt%)	< 0.1	0.1~1	1~10	10~80	> 80
	Spontaneous neutron generation rate	(²⁴⁰ Pu+ ²⁴² Pu)/Pu (wt%)	< 1	1~10	10~20	20~50	> 50
Material quantity	Mass of an item (kg)		10	10~100	100~500	500~1000	> 1000
	No. of items for SQ		1	1~10	10~50	50~100	> 100
	No. of SQ (material Stock or flow)		> 100	50 ~ 100	10 ~ 50	10 ~ 1	< 1
Material form	Chemical/physical form	U	Metal	Oxide/ Solution	U compounds	Spent fuel	Waste
		Pu	Metal	Oxide/ Solution	Pu compounds	Spent fuel	Waste
		Thorium	Metal	Oxide/ Solution	Th compounds	Spent fuel	Waste
Nuclear technology	Enrichment		Yes			No	
	Extraction of fissile material		Yes			No	
	Irradiation capability of target		Yes			No	

- User Requirement 1.2

□ Accountability

- ✓ MUF: 0
- ✓ Measurement method/equipment

The item accounting for DUPIC spent fuel shall be applied at the disposal facility. The weighing and NDA systems for the bundle accounting would be applied at the disposal facility.

□ Applicability of C/S measures

- ✓ The contaminant and surveillance systems can be easily installed at the disposal facility.
- Applicability of other Monitoring systems
 - ✓ Feed material measurement: DUPIC spent fuel bundle scanning system
- Detectability of nuclear material
 - ✓ The DUPIC spent fuel are easily identified by NDA.
 - ✓ The radiation signature gradually becomes weaker with time, and after hundreds of years, the radiation field will be no longer active.
 - ✓ The DUPIC spent fuels in the disposal facility are easily identified by the passive system.
- Difficulty to modify the process
 - ✓ Extent of automation
 - Fuel loading and handling systems are not fully automated.
 - ✓ Availability of data
 - The data is on-line monitored by the operator.
 - ✓ Authenticability of data
 - The data is very authentic because all the activities in the plant are open to IAEA.
 - ✓ Transparency of process
- Difficulty to modify facility design
 - ✓ Verifiability of facility design by inspectors

Table 39. Evaluation of URPR1.2 of Step D8

Indicators	Evaluation Parameter		Evaluation scale				
			W			S	
			VW	W	M	S	VS
Accountability	MUF/SQ	Kg Pu or ²³³ U	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with HEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Kg ²³⁵ U with LEU	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
		Ton Th	> 2	2 ~ 1	1 ~ 0.5	0.5 ~ 0.1	< 0.1
	NDA measurement capability by inspectors						
Applicability of C/S measures	Applicability of containment measures		No			Yes	
	Applicability of surveillance measures		No			Yes	
	Applicability of other monitoring systems		No			Yes	
Detectability of nuclear material	Possibility to identify nuclear material by NDA		No			Yes	
	Hardness of radiation signature		No reliable signature			Reliable signature	
	Need for passive/active mode		Active			Passive	
Difficulty to modify the process	Extent of automation		N/A	Manual operation	N/A	Partial automation	Full automation
	Availability of data for inspectors		Very low	Low	Medium	High	Very high
	Authenticability of data to be provided for safeguards purpose		No			Yes	
	Transparency of process		No			Yes	
	Accessibility of material to inspectors for verification		No			Yes	
Difficulty to modify facility design	Verifiability of facility design by inspectors		No			Yes	

7. Conclusions

While the proliferation resistance evaluation methodology including Basic Principles, User Requirements and Indicator, etc. presented in the IAEA-TECDOC-1362 are comprehensive and very informative, the deficiency such as the redundancy and the lack of correspondence was complemented through the DUPIC Case Study during the INPRO Phase 1B, Part 1.

The revised evaluation methodology in the proliferation resistance area presented in the IAEA-TECDOC-1434 is a big step forward to assess the degree of the proliferation resistance of the nuclear energy system quantitatively.

As performing this Extended Case Study on the whole DUPIC fuel cycle, the evaluation methodology of IAEA-TECDOC-1434 is further modified, and the modified methodology is applied to the whole DUPIC fuel cycle.

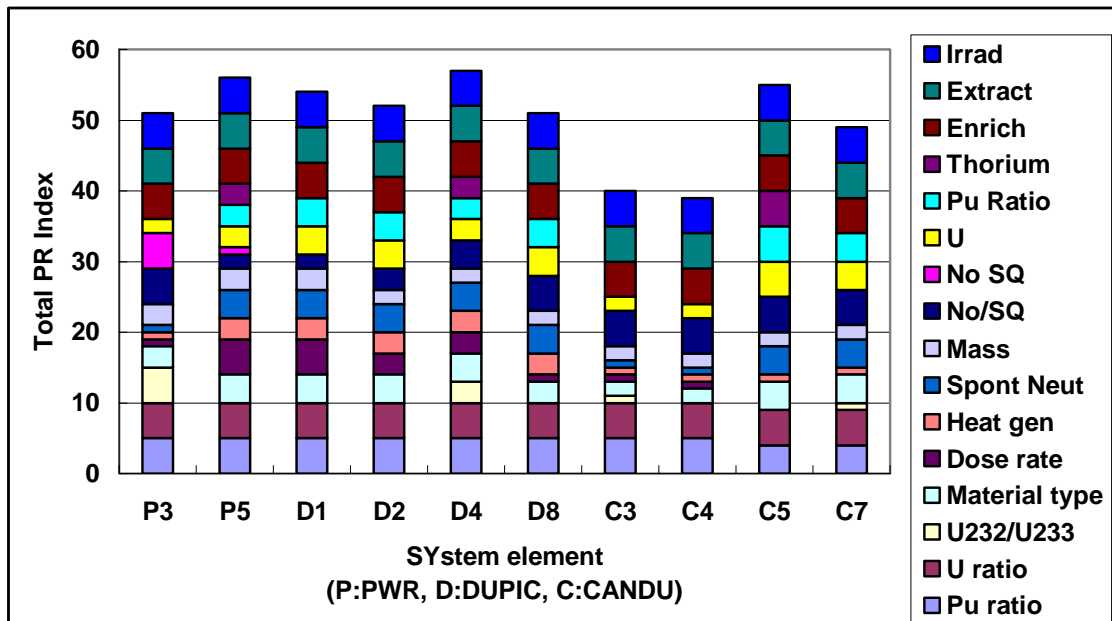


Figure 22. Aggregation of UR1.1 assessment results of the system elements

And, the assessment results of UR1.1 for the whole fuel cycle of DUPIC can be aggregated as shown in Figure 22 if we assumed that the quantified evaluation scores are classified into from 1 to 5 as an equal interval and significance. It is noted that the higher the column is in Figure 22, the more resistant the proliferation is. For example,

the total PR Index of C4 (Transportation of new CANDU fuel to CANDU plant) is the lowest among the system elements which are considered at the present study, while the total PR Index of D4 (Interim storage of spent DUPIC fuel) is the highest. It stands for that the PR of the system element of D4 is the highest while that of the system element of C4 is the lowest.

The Basic Principle 1 of the modified methodology is mainly dealing with the intrinsic features and extrinsic measures of the proliferation resistance evaluation, which is rather clear to understand and utilize them to the innovative nuclear energy system. However, the selection of the reliable and commonly acceptable values of the acceptance limits is still needed further study.

The Basic Principle 2 of the modified methodology is mainly dealing with the multiplicity and robustness of the barriers against the proliferation. The evaluation of the multiple barriers and its robustness requires the establishment of the comprehensive diversion scenarios for the proliferation, which is difficult to determine in case of the future innovative nuclear energy system.

Regarding the cost effectiveness, it is more difficult to evaluate the proliferation resistance quantitatively especially in case of the innovative nuclear system because of the lack of the detailed design information.

The integration of the evaluation results and the effective presentation of the evaluation results for the designers and policy makers are another area which needs further study.

The observation and recommendations made as the results of the extended case study on the whole DUPIC fuel cycle can be utilized for preparing the User Manual in the future to provide the stepwise evaluation method of proliferation resistance to the Member States of IAEA.

Moreover, the new modified INPRO PR methodology proposed by Korean Extended Case Study is still open to the improvement in further developing PR evaluation methodology.

ACKNOWLEDGEMENT

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External Experts:

Mr. R. Nishimura, AECL, Canada (Chairman)
Mr. G. Cojazzi, JRC, Italy
Ms. M. Lesage, AREVA, France
Mr. A. Kumar, BARC, India
Mr. J.H. Park, KAERI, Republic of Korea
Mr. G. Pshakin, IPPE, Russia
Mr. G. Stein, FZ-Juelich, Germany
Mr. M.S. Yang, KAERI, Republic of Korea
Mr. M. Zentner, PNNL, U.S.A.
Mr. R. Verslius, DOE, U.S.A. (Attended partially on the third day)

IAEA Internal Experts:

Mr. A. Rao, SH-NPTDS & INPRO Coordinator
Mr. E. Haas, SGCP
Mr. S.C. Kim, NSNI
Mr. F. Depisch, INPRO ICG
Mr. Y. Bussurin, INPRO ICG
Mr. M. Moriwaki, NENP/INPRO ICG
Mr. P. Villalibre, INPRO ICG
Mr. V. Usanov, INPRO ICG
Mr. J. Kim, INPRO ICG (Scientific Secretary)

References

1. “Guidance for the evaluation of innovative nuclear reactors and fuel cycles”, Report of Phase 1A of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), IAEA-TECDOC-1362, June 2003.
2. “Methodology for the assessment of innovative nuclear reactors and fuel cycles,” Report of Phase 1B, Part 1 of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), IAEA-TECDOC-1434, December 2004.
3. H. Choi, M.S. Yang, et al., “Progress of the DUPIC fuel compatibility analysis-1 : Reactor physics”, *Nuclear Technology*, Vol. 153, January 2006
4. A.G. Croff, “ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials,” *Nuclear Technology*, Vol. 62, 1983.
5. W.I. Ko, et al., “Proliferation Resistance of the Lithium Reduction Process” *Proceedings of the Korean Nuclear Society Spring Meeting*, Kwangju, Korea, May 2002.
6. “IAEA Safeguards Glossary 2001 Edition,” International Nuclear Verification Series, 2002.
7. M.S. Yang, “Korean National Case Study on DUPIC fuel cycle”, KAERI, Korea, Submitted to IAEA, June 2004

APPENDIX

Review Comments by IAEA INPRO PR Consultancy Meeting

INPRO Consultancy Meeting on results of the Korean assessment study and the draft proliferation resistance chapter of the INPRO Manual was held at IAEA VIC (F0579) from April 11 to 13, 2006. Review comments were proposed during CM meeting and described as followings. All the comments will be reflected in the preparation of INPRO PR Manual based on the results of Korean assessment study.

1. Review of Basic Principles and User Requirements

The new structure of the Indicators proposed in the Korean Case Study was reviewed. Comments arising in the discussion are summarized below.

- **BPPR1 and BPPR2:** The consultants accepted this UR with no changes or comments.
 - **URPR1.1:** In response to discussion on the addition of the concept of “nuclear technology”, Mr. Haas clarified that the concept was added to cover the impact of these technologies on the PR of the fuel cycle as a whole when an INS is misused to generate nuclear material for a nuclear explosive. The consultants agreed that while URPR1.2 covers misuse of the nuclear technology in a facility within an INS, the inclusion of the term “nuclear technology” in URPR1.1 is necessary in considering the proliferation resistance of an entire fuel cycle, and covers misuse through such means as technology transfer or clandestine replication of equipment/facilities.
 - **URPR1.2:** The consultants accepted this UR with no changes or comments.
 - **URPR1.3:** Mr Cojazzi commented that URPR1.3 could be appropriately linked to either BPPR1 or BPPR2. After discussion, the consultants agreed that URPR1.3 should remain associated with BPPR1.
 - ✓ Mssrs Cojazzi and Zentner noted that unlike URPR1.1 and URPR1.2, User Requirement URPR1.3 is country-specific and only performed once for all facilities within a state. Discussion ensued as to whether this should be explicitly recognized, for example by restoring it to being the first UR in the

list. The consensus of the consultants was that although URPR1.3 is different, recognition of that difference is not essential to an INPRO PR assessment.

- ✓ Mr Stein suggested, and the consultants agreed that in addition to states' commitments, obligations and policies regarding non-proliferation, URPR1.3 should be expanded to include the states' proliferation history. The consultants agreed that the phrase "to fulfill international standards" should be deleted from URPR1.3.
- ✓ URPR2.1, URPR2.2: The consultants accepted these URs with no changes or comments.

2. Review of Indicators and Evaluation Parameters

The consultants agreed that the scales shown in the DUPIC Case Study and draft manual are examples and subject to further consideration. The consultants recommend that this should be clearly stated in the Manual.

The consultants recommend that the Manual contain detailed definition, explanation and justification for each of the Evaluation Parameters and Indicators. For example the importance of the Evaluation Parameters associated with the Material Quantity Indicator is better understood after reading the example in the case study; it should be evident from the definition/explanation of the EP (Evaluation Parameter).

The consultants recommend that the Manual contain detailed explanation of the example scale for each Evaluation Parameter. For example, the term "waste" used in the example scale for the evaluation parameters associated with Indicator INPR1.1.3, Material Form requires clear definition.

The consultants recommend that the Manual clearly explain that some of the evaluation parameters are interdependent. e.g., material form and isotopic composition; isotopic composition and spontaneous neutron generation rate. Mr. Haas noted that care is required to avoid double counting if inter-related Evaluation Parameters or Indicators are aggregated.

- **INPR1.1.1:** The consultants agreed that some of the EPs associated with INPR1.1.1 are interdependent but that there may be value in the context of an INPRO assessment, in showing these redundant values. The consultants accepted that the

term “material attractiveness” reflects a judgement and is used in the definition of a UR, and that the term “material quality” is more objective and therefore the appropriate term for the Indicator 1.1.1. The tables contained in the documentation provided to the consultants could be clarified by writing “material attractiveness” in the title next to URPR1.1. Other comments arising during discussion of this Indicator and its Evaluation Parameters were later generalized and are covered by the preceding points in this section of the minutes.

- **INPR1.1.2:** In response to initial discussion, Mr. Haas reviewed the rationale for the Evaluation Parameters associated with Indicator 1.1.1 (Material Quantity). The consultants accepted this Indicator and its associated Evaluation Parameters with no changes or comments.
- **INPR1.1.3:** The consultants accepted this Indicator and its associated Evaluation Parameters with no changes or comments.
- **INPR1.1.4:** Mr. Cojazzi noted that the scale for the Evaluation Parameters associated with INPR1.1.4 Nuclear Technology, were too coarse because they do not discriminate between systems and do not reflect differences in the difficulty of misuse. For example, any reactor can be used to irradiate targets but the difficulty of doing so can vary from one to the next. The consultants agreed that the scales for the Evaluation Parameters associated with the Nuclear Technology Indicator require finer gradation. The scales should take into account the capability of the technology (e.g. lab scale, pilot plant, demonstration, etc.) and the technical know-how that it provides.
 - In response to questions about the inclusion of hot cells as a possible Evaluation Parameter, Mr Haas clarified that the indicator and its evaluation parameters is intended to address facilities that could be used directly for production of weapons material.
- **INPR1.2.1:** Discussion clarified that the example “MUF/SQ” scales proposed in the DUPIC Case Study are dimensionless and that the MUF should be Σ -MUF or LEMUF. e.g. Σ -MUF in kg divided by SQ in kg. The consultants recommend that the IAEA Safeguards Department clarify whether Σ -MUF or LEMUF is the appropriate term.
 - The consultants further recommend that “ Σ -MUF/SQ” is not an appropriate Evaluation Parameter for thorium, natural, or depleted uranium. The reference to “MUF/SQ for Th” should be removed from the example scales in the DUPIC Case Study.

- The consultants recommend that Table 3 on page 53 of the IAEA Safeguards Glossary should form the basis for the scales for the Evaluation Parameters associated with Indicator INPR1.2.1, Accountability.
- The consultants recommend that further work is required to clarify the intent of the Evaluation Parameter called “NDA measurement capability by inspectors” and to define a meaningful scale. The consultants believe that the intent of this Evaluation Parameter is to consider the ability of the inspector to quantitatively verify the accountancy declarations of the operator through independent NDA or DA. Based on this understanding, the consultants recommend that the Evaluation Parameter should be renamed “inspectors measurement capabilities” to recognize DA.
- **INPR1.2.2:** The consultants recommend that the name of this indicator should be changed to “Amenability for C/S and monitoring systems”. The Evaluation Parameters should be changed to use the term “Amenability” rather than “Applicability”. Evaluation of the Evaluation Parameters for this indicator requires analysis of detailed acquisition paths because C/S measures are related to elements rather than facilities within an INS. The consultants recommend that INPRO consider expanded scales for the Evaluation Parameters, that reflect a range of effectiveness and efficiency, and that are illustrated with examples.
 - The consultants recommend that the term “monitoring” used in the third Evaluation Parameter should be clearly defined. It is not clear if this term includes unattended/remote monitoring, motion sensors, automated measurement systems, NRTA systems, process monitoring, or other forms of monitoring.
- **INPR1.2.3:** This indicator pertains to the declared material in an INS and not to the detectability of undeclared material associated with misuse involving an INS. The consultants recommend that a new indicator called “detectability of misuse” should be associated with URPR1.2, along with appropriate Evaluation Parameters. Some examples of situations that should be covered by this indicator are: overproduction using undeclared material, the presence of nuclear materials that should not appear in a system element, and undeclared modifications to a facility.
 - The consultants recommend that the second Evaluation Parameter be reworded to replace the term “hardness” with “detectability” to reflect that this EP refers to how easily the radiation signature can be detected taking into account the possible use of shielding to conceal diversion, the background noise in the signals, etc.

- The consultants recommend that the third Evaluation Parameter, “Need for active/passive mode” be deleted.
- **IRPR1.2.4:** The consultants recommend that the third Evaluation Parameter, Authenticability of data, should be deleted because it cannot be evaluated early in the design process.
 - The consultants recommend that the Manual clarify that the second Evaluation Parameter, “Availability of data for inspectors,” refers to Near Real Time Accounting (NRTA). The consultants further recommend that the scale for “Availability of data for inspectors” be considered further; it is related to the willingness of the operator/state to implement a NRTA system and share the data.
- **INPR1.2.5:** The consultants accepted this Indicator and its associated Evaluation Parameters with no changes or comments.
- **INPR2.2.1:** Mssrs Stein and Haas noted that with Integrated Safeguards, the focus of verification is shifting to include a combination of state-level and facility-level safeguards. The activities for facility-level safeguards can be reduced when state evaluations are taken into account and this can affect the cost analysis associated with INPR2.2.1.
- **INPR1.3.1:** The consultants recommend that a new Evaluation Parameter 1.3.1.x named “Recorded violations of non-proliferation commitments” be added. A possible scale for this new EP would be yes, no. The consultants recommend that the manual state that this EP be evaluated first.
 - The Evaluation Parameters should be changed as follows:
 1. Party to the NPT,
 2. Safeguards Agreements according to the NPT in force,
 3. For those who are not party to the NPT, other safeguards agreements (e.g. INFCIRC/66) in force,
 4. Additional Protocols in force,
 5. Export Control...
 6. Regional SSAC...
 7. State SSAC...
 8. Relevant International Conventions...
 9. NWFZ treaties...
 10. Verification Approach...
 11. Recorded violations of non-proliferation commitments
 - With regard to relevant international conventions, a list of such conventions relevant to PR should be included in the assessment.

- The consultants recommend that some Evaluation Parameters, such as Nuclear Weapons Free Zone Treaties, State SAC, Regional SAC, Relevant International Conventions, should have a scale that includes Not Applicable. The manual should clearly state that NA is only for EPs that may not be relevant because the treaty or commitment is not available for the country being assessed.
- The consultants recommend that the Evaluation Parameter (shown as EPPR1.3.1.8 in the DUPIC Case Study) pertaining to an agreed-to verification approach be moved to become INPR2.1.3.
- **INPR1.3.2:** The indicator should be renamed “Institutional Structural Arrangements”. The consultants clarified that international dependency pertains to dependence on another state for uranium/thorium resource.
- **INPR2.1.1, INPR2.1.2:** The consultants recommend that the formulation for INPR2.1.1 and INPR2.1.2 should be revised to read “The extent by which an INS is covered by multiple intrinsic features, and extrinsic measures” and should delete the second sentence about the “fraction of plausible acquisition paths.” The Evaluation Parameters shown in the Korean Case Study along with the proposed scales for these indicators should not be adopted. The purpose of these indicators is to encourage designers to incorporate intrinsic features. The proposed formulation for INPR2.1.1 and 2.1.2 are shown below.
 - INPR2.1.1 The extent by which the INS is covered by multiple intrinsic features and extrinsic measures*
 - INPR2.1.2 Robustness of barriers covering each acquisition path.*
 - *The evaluation of these Indicators would require detailed pathway analysis
 - The consultants recommend that the Manual should state that the evaluation of these Indicators would require detailed pathway analysis. This statement should also be incorporated into the formulation of the User Requirement or the Indicators.
 - The consultants recommend that the Manual should clearly state that intrinsic features should be included in an INS to decrease the impact of safeguards implementation and verification on the facility.
- **INPR2.2.1:** The consultants discussed what costs should be included in the cost analysis. The consultants noted that it may be very difficult to separate the costs associated with PR from the basic costs for an INS. Mr. Haas clarified that the original intent of cost in the context of INPR2.2.1 is to only consider the costs for extrinsic measures and the costs for additional intrinsic features added to the basic

INS design to provide PR. The latter would include costs resulting changes made to the INS as a result of a PR analysis.

- The consultants recommend that the indicator should be replaced with either one of the following two proposed formulations:
- INPR2.2.1 (1st proposed formulation): *Analysis has been provided by the designer showing that cost effective features have been employed (taking into consideration a balance between facility and verification costs)*. A scale for this indicator could be: Yes analysis is done; Analysis is going to be done; Analysis is not going to be done. The acceptance limit for this scale would be Yes analysis is done. Analysis is going to be done would indicate that this aspect of the INPRO assessment cannot be completed at this time.
- INPR2.2.1 (2nd proposed formulation): Cost to incorporate those intrinsic features and extrinsic measures, which are required to provide PR.*

*The evaluation of this Indicator would require cost analysis

- **INPR2.2.2:** The consensus of the consultants was that Indicator INPR2.2.2 contained in IAEA TECDOC 1434 should be retained as INPR2.2.2 despite a proposal by the Korean Case Study team to move this to being an Evaluation Parameter under INPR1.3.1 regarding States' commitments and obligations. The consultants recommend that the Manual should explain that INPR2.2.2 is intended to show that the costs are acceptable to both the verification authority and the State/facility.